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DTE Energy

10CFR50.90
10CFR50.67

February 13, 2003
NRC-03-0007

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

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

Subject: Proposed License Amendment for the Implementation of Alternative Radiological Source Term Methodology

Pursuant to 10 CFR 50.90 and 10 CFR 50.67, Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License NPF-43, Appendix A, Technical Specifications (TS) based on a re-evaluation of the Loss of Coolant Accident (LOCA) Design Basis Accident (DBA) using the Alternative Source Term (AST) methodology. 10 CFR 50.67 provides a mechanism for operating license holders to utilize AST in the evaluation of the impact of proposed changes on affected design-basis radiological analyses. Regulatory guidance for the implementation of AST is provided in Regulatory Guide (RG) 1.183 and Standard Review Plan (SRP) Section 15.0.1.

In support of the proposed changes, Detroit Edison has performed a radiological consequence analysis of the DBA LOCA based on the guidance in RG 1.183 and SRP Section 15.0.1.

Detroit Edison has previously implemented two license amendments related to limited scope application of AST. The first amendment (No. 143) is related to the revision of fission product release timing for DBAs and the second amendment (No. 144) is for the re-evaluation of the Fuel Handling Accident (FHA) using AST. This application does not affect changes requested in these two previous license amendments. However, this submittal reflects an update of the input data used in the supporting analysis of the FHA radiological consequences to provide consistency with input used for the LOCA analysis.

Similar license amendments for the implementation of AST have been previously approved by the NRC for other nuclear plants such as for the Hope Creek Generating

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Station (ADAMS Accession No. ML012600176) and the Surry Power Station, Units 1 and 2 (ADAMS Accession No. ML020710159).

A list of enclosures to this letter is provided below:

- Enclosure 1 provides a description and a safety analysis of the proposed changes.
- Enclosure 2 provides a compliance matrix demonstrating compliance with RG 1.183.
- Enclosure 3 provides an analysis of the issue of Significant Hazards Consideration using the standards of 10 CFR 50.92.
- Enclosure 4 provides information supporting Environmental Considerations per 10 CFR 51.22.
- Enclosure 5 provides marked up pages of the existing TS to show the proposed changes and a typed version of the affected TS pages with the proposed changes incorporated.
- Enclosure 6 provides marked up pages of the existing TS Bases showing the proposed changes (for information only).

Detroit Edison has concluded that changes proposed in this submittal do not result in a significant hazards consideration. The changes have also been evaluated using the identification criteria for licensing and regulatory actions requiring an environmental assessment as specified in 10 CFR 51.21. The proposed changes meet the eligibility criteria for a categorical exclusion as set forth in 10 CFR 51.22; therefore, an environmental assessment of the proposed changes is not required.

Detroit Edison requests NRC approval of this license amendment by September 30, 2003, with an implementation period of within 60 days following NRC approval.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

William O'Connor

Enclosures

cc: M. A. Ring
J. F. Stang, Jr.
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, WILLIAM T. O'CONNOR, JR., do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

WILLIAM T. O'CONNOR, JR.
Vice President - Nuclear Generation

On this 13th day of February, 2003 before me personally appeared William T. O'Connor, Jr., being first duly sworn and says that he executed the foregoing as his free act and deed.

Karen M Reed
Notary Public

KAREN M. REED
Notary Public, Monroe County, MI
My Commission Expires 09/02/2005

NRC-03-0007
ENCLOSURE 1

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR
FULL SCOPE IMPLEMENTATION OF
ALTERNATIVE SOURCE TERM

DESCRIPTION AND SAFETY ANALYSIS
OF THE PROPOSED CHANGES

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

A. SUMMARY OF THE PROPOSED CHANGES

In accordance with 10 CFR 50.67, "Accident source term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Detroit Edison requests changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-43 for Fermi 2. The proposed changes are based on the implementation of the Alternative Source Term (AST) methodology; however, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 1) will continue to be used as the radiation dose basis for equipment qualification.

On December 23, 1999, the NRC published regulation 10 CFR 50.67 in the Federal Register. This regulation provides a mechanism for operating license holders to revise the current accident source term used in design-basis radiological analyses with an AST. Regulatory guidance for the application of AST is provided in Regulatory Guide 1.183 (Reference 2) and Standard Review Plan, Section 15.0.1 (Reference 3).

Detroit Edison has performed a radiological consequence analysis of the Loss of Coolant Accident (LOCA) Design Basis Accident (DBA) based on the guidance provided in References 2 and 3.

Detroit Edison has previously received license amendments related to limited scope application of AST for the revision of fission product release timing for DBAs (Reference 4) and for the re-evaluation of the Fuel Handling Accident (FHA) using AST (Reference 5). This application does not affect changes requested in these two previous license amendments. However, the supporting analysis of the consequences of the FHA has been updated mainly in two areas. First, core source term data has been updated using the ORIGEN-S (Reference 7) computer code. Second, offsite atmospheric relative concentrations (χ/Q's) have been reevaluated considering offshore wind directions. The original design basis atmospheric relative concentrations excluded consideration of offshore wind directions. Considering all wind directions simplifies the analysis and results in more conservative χ/Q values.

The proposed changes to the current licensing basis for Fermi 2, that are justified by the AST analysis of the DBA LOCA, are described below. These changes will result in revisions to the Technical Specifications (TS), TS Bases, and other licensing basis documents.

Specific changes include:

- Revision of TS Surveillance Requirement (SR) 3.6.1.3.11 to increase allowed equivalent secondary containment bypass leakage to less than or equal to 0.05 times the maximum allowable leakage rate for the primary containment (L_a). The TS Bases are revised to

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indicate that the allowable bypass leakage limit may be applied after adjusting the measured leak rates for piping deposition credit determined using the RADTRAD computer code Brockmann-Bixler correlations (Reference 8).

- Revision of SR 3.6.1.3.12 to increase main steam isolation valve allowable leakage to 100 SCFH per steam line and a total of 250 SCFH for all four lines when tested at greater than or equal to 25 PSIG.
- Deletion of Main Steam Isolation Valve (MSIV) Leakage Control System (LCS) TS 3.6.1.9, since this system is no longer credited in any accident analysis.
- Revision of SR 3.6.4.1.5 to increase the allowed secondary containment draw down time from 567 seconds (9 minutes and 27 seconds) to 12 minutes.
- Revision to the Control Room Envelope (CRE) design basis assumptions to increase postulated unfiltered inleakage into the CRE to 900 SCFM.
- Development of new offsite and control room atmospheric relative concentrations (χ/Q's) using the site-specific meteorology data collected between 1995 and 1999.

The proposed changes are further described in Section E of this Enclosure. The marked-up TS pages and re-typed pages are provided in Enclosure 5. Marked up TS Bases pages reflecting corresponding changes are provided in Enclosure 6.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

B.1 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

SR 3.6.1.3.11 requires the verification that the combined leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system to be $\leq 0.04\ L_a$ when pressurized to $\geq$ 56.5 PSIG.

SR 3.6.1.3.12 requires verification that the combined leakage rate for all MSIV leakage paths is $\leq$ 100 SCFH when tested at $\geq$ 25 PSIG.

B.2 TS Section 3.6.1.9 "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)"

TS Section 3.6.1.9 requires that two MSIV LCS subsystems be operable in Modes 1, 2, and 3, and includes associated surveillance requirements.

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B.3 TS Section 3.6.4.1, "Secondary Containment"

SR 3.6.4.1.5 requires verification that each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 567 seconds.

C. BASES FOR THE CURRENT REQUIREMENTS

C.1 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. The limit of 0.04 L_a specified in SR 3.6.1.3.11 for secondary containment bypass leakage addresses leakage paths with bypass potential identified in Table 6.2-2 of the Updated Final Safety Analysis Report (UFSAR), with the exception of steam lines, and assures conformance with design basis accident radiological evaluation, for unfiltered containment leakage. The PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident, thus providing assurance that primary containment will function as assumed in the safety analyses. The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs are minimized.

SR 3.6.1.3.12 leakage rate limit of ≤ 100 SCFH combined for all four main steam lines is to ensure that the MSIV LCS has the ability to provide a positive pressure seal between MSIVs. This leak test is performed in lieu of 10 CFR 50, Appendix J, Type C test, based on an exemption to 10 CFR 50, Appendix J testing. As such, this leakage is not combined with the Type B and C leakage rate totals.

C.2 TS Section 3.6.1.9, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)"

The MSIV LCS supplements the isolating function of the MSIVs after a DBA LOCA. The MSIV LCS is designed to operate at a higher pressure than the primary containment after a postulated LOCA to provide protection against potential leakage of radioactive contaminants to the environment.

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C.3 TS Section 3.6.4.1, "Secondary Containment"

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be operable, or that take place outside primary containment.

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.5 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 567 seconds. The 567-second requirement accounts for 33 second SGT System startup delay time such that the system function is assured to be available within 10 minutes consistent with the plant design basis. This cannot be accomplished if the secondary containment boundary is not intact.

D. NEED FOR REVISION OF THE REQUIREMENTS

The proposed changes to the TS will allow Fermi 2 to apply the results of the plant-specific AST LOCA analysis using the guidance in Reference 2 and meeting the requirements of 10 CFR 50.67. Approval of these changes will provide a more realistic source term for Fermi 2 that will result in a more accurate assessment of the limiting DBA radiological doses. This allows relaxation of some current licensing basis requirements as described in Section E of this Enclosure. The AST methodology can also be used to support future evaluations and license amendments.

The increases in limits on secondary containment bypass leakage and MSIV leakage, and the elimination of the requirements for MSIV LCS operation will result in a reduction in personnel exposure due to valve maintenance being performed on the PCIVs, and maintenance of the MSIV LCS. In addition, eliminating the MSIV LCS function will reduce the demand requirements on the Non-Interruptible Air Supply (NIAS); thereby increasing the margins for operation of this safety system.

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Furthermore, the increase in postulated control room unfiltered inleakage in the analysis provides additional margin in establishing conformance to the design basis assumption for this parameter.

E. DESCRIPTION OF THE PROPOSED CHANGES

E.1 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The proposed change revises SR 3.6.1.3.11 to increase the combined leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system from ≤ 0.04 L_a to ≤ 0.05 L_a when pressurized to ≥ 56.5 psig. The proposed change also revises the term "leakage rate" to "equivalent leakage rate" to indicate that piping deposition credit may be taken in determining the leakage rate.

The proposed change revises SR 3.6.1.3.12 to increase the combined MSIV leakage rate for all four main steam lines from ≤ 100 scfh to ≤ 250 scfh and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig.

The bases for these changes are as follows:

The existing exemption from 10 CFR 50, Appendix J, Type C test requirements for MSIVs is requested to be continued but on another basis. The current basis is because the MSIV LCS prevents release of this leakage to the environment. The new basis is that the MSIVs have separate leakage limits in SR 3.6.1.3.12, and the dose consequences of this leakage path are evaluated separately and added to those calculated from primary containment L_a leakage, including secondary containment bypass leakage. As such, MSIV leakage is not combined with Type B and C leakage rate totals.

The increase in allowable MSIV leakage has been assumed in the DBA LOCA analysis to ensure that regulatory dose limits are met. This proposed change provides additional margin and reduces personnel doses associated with MSIV valve maintenance. Leakage measurement at greater or equal to 25 PSIG pressure continues to be acceptable because of expected containment pressures, and expected holdup in main steam piping.

The MSIV piping leakage has been evaluated with the credit allowed under AST analyses for deposition of aerosols, elemental iodine and organic iodine in qualified piping, as well as holdup in transit.

NRC approval for similar deposition credit for other bypass pathways is also requested, using either one of the following evaluation approaches. For both of these

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approaches, deposition credit only applies to seismically qualified piping or to piping within secondary containment.

Approach A: Treatment as Contributing to Secondary Containment Bypass Leakage

1. For a particular penetration with secondary containment bypass leakage potential, analyze dose implications if the entire bypass limit of 5% L_a passes through that penetration, using RADTRAD computer code, with and without piping deposition credit. The Brockmann-Bixler option must be used for establishing pipe deposition effects.
2. Take the ratio of [dose with deposition credit TEDE results] to [dose without deposition credit TEDE results] for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR).
3. Select the highest ratio, but not less than 0.01 (SGT System filter equivalent).
4. Use this factor to adjust measured leak rate before totaling with other Type B and C leakage for secondary containment bypass leakage surveillance. The before-adjustment leak rate of any penetration is not allowed to exceed 5% L_a.

A Feedwater penetration was evaluated as an example with Approach A, with the result that the measured penetration leak rate can be multiplied by 0.0721 before Type B and C totaling and comparison against 5% L_a.

Approach B: Treatment as Contributing to Standby Gas Treatment (SGT) System Filtered Leakage

1. For a particular penetration with secondary containment leakage bypass potential, analyze dose implication if the entire non-bypass flow (95% L_a) passes through that penetration, using RADTRAD with piping deposition credit but without SGT System credit, and, as a separate case, with SGT System credit but without piping deposition credit.
2. Take the ratio of [dose with deposition credit TEDE results] to [dose with SGT System TEDE results] for EAB, LPZ, and CR.
3. Select the highest ratio.
4. Use this factor to adjust measured leak rate before totaling with other Type B and C leakage for total containment leakage (not secondary containment bypass leakage). The before-adjustment leak rate is not allowed to exceed 95% L_a

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The same Feedwater penetration was again evaluated as an example with Approach B, with the result that the measured penetration leak rate can be multiplied by 3.25 before Type B and C totaling and comparison against L_a

E.2 TS Section 3.6.1.9, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)"

The proposed change deletes this section of the TS because this system is no longer credited in the DBA safety analyses.

E.3 TS Section 3.6.4.1, "Secondary Containment"

The proposed change revises SR 3.6.4.1.5 to increase the time limit required for secondary containment draw down using the SGT System from 567 seconds (9 minutes and 27 seconds) to 12 minutes.

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F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

1.0 Introduction

The fission product release from the reactor core into containment is referred to as the "source term" and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of TID-14844 (Reference 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island. NUREG-1465 (Reference 6) was published in 1995 with revised AST for use in the licensing of future Light Water Reactors (LWRs). The NRC, in 10 CFR 50.67, later allowed the use of the AST described in NUREG-1465 for operating plants. This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

The five release phases representing the progress of a severe accident in a LWR are described in NUREG-1465 as:

1. Coolant Activity Release
2. Gap Activity Release
3. Early In-Vessel Release
4. Ex-Vessel Release
5. Late In-Vessel Release

Current DBA evaluations considered phases 1, 2, and 3 and assumed these releases would occur at time zero. Phases 4 and 5 are related to severe accident evaluations and are not addressed in DBA analyses. Under the AST, the coolant activity release is assumed to occur instantaneously and end with the onset of the gap activity release. Early In-Vessel release is assumed to start at the end of the gap activity release period. Phases 4 and 5 continue to be addressed in existing severe accident evaluations and procedures.

This proposed license amendment involves a full-scope application of the AST, addressing the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release as described in Reference 2.

Detroit Edison has performed a radiological consequence analysis of the limiting DBA that results in an offsite exposure (i.e., LOCA). Additionally, the AST analysis of the FHA has been updated to maintain consistency with the new offsite

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atmospheric relative concentrations (χ/Q's) and core radionuclide inventory associated with the DBA LOCA AST analysis. The LOCA AST analysis was performed to support the proposed changes in this license amendment. The AST analyses have been performed in accordance with the guidance in References 2 and 3. The implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the DBA LOCA and FHA that could potentially result in control room and offsite doses,
- Analysis of the atmospheric dispersion for the radiological propagation pathways, using methods endorsed by Reference 2,
- Calculation of fission product deposition rates, transport and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses, and
- Evaluation of suppression pool pH to ensure that the iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Reference 2.

2.0 Evaluation

2.1 The only DBA accident impacted by the proposed changes is the DBA LOCA. The LOCA analysis is documented in the Fermi 2 UFSAR, Section 15.6.5. In the revised LOCA analysis, the control room and offsite doses were calculated using methods and input assumptions consistent with the AST methodology. The analysis was performed in accordance with Reference 2 to confirm compliance with the acceptance criteria presented in 10 CFR 50.67. In addition, the FHA evaluation described in section 15.7.4 of the UFSAR is updated using consistent core fission product inventory and χ/Q's used in the LOCA analysis.

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2.2 Environmental Qualification

Detroit Edison has determined that continued compliance will be maintained with NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations." The source term associated with environmental qualification of equipment will remain consistent with previous commitments under 10 CFR 50.49. This is consistent with the NRC guidance provided in Reference 19. AST post-accident dose rates to vital areas used in post-accident operations have been re-evaluated and these areas were found to remain accessible with analyses and design changes associated with the AST.

2.3 Fission Product Inventory

The ORIGEN-S computer code (Reference 7) was used as the basis of a calculation of the fission product source term inventories. The inventories were determined based on the licensed core power level (i.e., 3430 megawatts thermal [MWt]) and further adjusted to 102% to 3499 MWt in support of the AST evaluations. The fission product inventory is primarily based on a core bundle average exposure of 35 GWD/MTU, the bounding average core exposure for a 24-month fuel cycle. The Co-58 and Co-60 source terms were adopted from the RADTRAD BWR default nuclide inventory file. The pre-accident core activities are provided in Table 1.

2.4 Radiological Consequences

New calculations were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the postulated LOCA and FHA scenarios.

The RADTRAD computer code (Reference 8) was used for these calculations. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. The RADTRAD code is a publicly available computer code and has been used in NRC safety reviews.

Offsite atmospheric relative concentrations (χ/Q's) were calculated using the guidance of Regulatory Guide 1.145 (Reference 9) and the PAVAN computer code (Reference 10). The code has been used in NRC safety reviews. Offsite χ/Q's have been reevaluated considering offshore wind directions. The original design basis atmospheric relative concentrations excluded consideration of offshore wind directions. Considering all wind directions simplifies the analysis and results in more conservative χ/Q values.

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Control room χ/Q's were calculated with the ARCON96 computer code (Reference 11). The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. The code has been used in NRC safety reviews. The application of the code is consistent with the methodology used in support of the FHA AST analysis approved by the NRC in Reference 5. Where applicable, the revised analysis credits χ/Q reductions for dual control room intakes.

Airborne radioactivity drawn into the control room envelope results in both internal and external dose components that are used in the TEDE dose calculation. The noble gas inventory within the control room is the main contributor to the gamma ray whole body (i.e., external) dose component of the TEDE; the non-noble gas radionuclides, principally iodines, contribute to the internal organ dose component via the inhalation pathway.

Dose contributions from radioactive sources outside of the control room were also considered, including airborne activity in primary and secondary containment, activity in the suppression pool, and activity accumulated on SGT System filters. The calculated dose contributions from these sources are lower than historically presented values due to additional analysis detail, as well as changes in release timing and reductions in filter loading resulting from AST assumptions.

3.0 Inputs and Assumptions

Release Mode

The accident analyses were performed for a core inventory based on 3499 MWt (102% of the current licensed power level of 3430 MWt) in accordance with Regulatory Guide 1.49 (Reference 14). The reactor core inventory for the analyses was primarily based on an assumed core bundle average exposure of 35 GWD/MTU, the bounding average core exposure for a 24-month fuel cycle.

Transport Mode

The meteorological measurements program at Fermi 2, used to determine the meteorological conditions prevailing at the site, meets the intent of Regulatory Guide 1.23 (Reference 16) and ANSI/ANS-2.5-1984 (Reference 15).

Recorded meteorological data are used to provide estimates of airborne concentrations of gaseous effluents and projected offsite radiation dose. Instrument calibrations and data consistency evaluations are performed routinely to ensure maximum data integrity. Better than 90 percent data recovery is attained from each measuring and recording system.

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Atmospheric relative concentrations (χ/Q's) were calculated, for the identified release paths, based on site-specific hourly meteorology data collected between January 1995 through December 1999. The relative concentrations developed represent a change from those used in the current UFSAR analyses. The values currently in the UFSAR are based on Regulatory Guide 1.3 (Reference 17). The AST analyses are based on zero velocity vent and ground level release χ/Q values calculated by the ARCON96 computer code for the control room, and ground level releases by the PAVAN code and Regulatory Guide 1.145 guidance for the EAB and LPZ boundaries.

The infiltration of unfiltered air into the control center complex occurs through three different paths: (1) the control center complex boundary, (2) the system components located outside the control center, and (3) backflow at the boundary doors as a result of ingress or egress to or from the control center.

During emergency filtration modes of operation, the control room ventilation system supplies outdoor air to maintain the control room at a minimum of 1/8-inch water column positive pressure with respect to the adjacent areas. Intentionally admitting outdoor air into the control center facilitates reduction of infiltration through the control center boundary by assuring that air is exfiltrating at an adequate velocity (i.e., a velocity through the control center boundary to develop and maintain a pressure of 1/8-inch water column.)

During the emergency filtration mode, air infiltration into the control center is assumed to be 900 CFM of outside air unfiltered inleakage.

The standard breathing rates used for control room personnel dose assessments and for the offsite personnel are shown in Table 2. Control room occupancy factors used in the analyses are also included in Table 2.

Removal Mode

Removal mechanisms are included in the applicable event-specific discussions below.

3.1 LOCA Inputs and Assumptions

The key inputs used in this analysis are included in Tables 3 through 6. These inputs and assumptions are grouped into three main categories (i.e., release, transport, and removal).

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LOCA Release Inputs

The LOCA analysis assumes the total primary containment leakage rate at the limit of 0.5 percent of primary containment air weight per day. The primary containment leakage is reduced by 50 percent after 24 hours, based on the post-LOCA drywell pressure history. Additionally, up to the equivalent of 5.0 percent of containment leakage (not including MSIV or ECCS leakage) is assumed to bypass the secondary containment and be released unfiltered to the atmosphere. The analysis assumes the maximum MSIV leakage rates of the proposed SR 3.6.1.3.12 to the environment. As discussed earlier, in Section E of this Enclosure, the proposed change revises SR 3.6.1.3.12 to increase the allowable limit for the combined leakage rate for all main steam lines to not exceed 250 scfh and an individual steam line leakage rate to not exceed 100 scfh, when tested at a pressure of not less than 25 psig. This leak rate is assumed to begin at the onset of the accident and to continue for 24 hours. The leak rate is assumed to be reduced by 50% after 24 hours and to continue at this rate throughout the 30-day duration of the postulated accident.

The DBA LOCA assumes that secondary containment draw down time with SGTS is 15 minutes from the onset of the gap release phase of the accident (i.e. about 17 minutes from the initiation time of the event, since gap activity release starts at 121 seconds.) The proposed revision to SR 3.6.4.1.5 (12 minutes) is conservative with respect to the draw down time assumed in the analysis. For releases from the SGT System, an effectively instantaneous release rate from the secondary containment is conservatively assumed.

The analysis assumes an ECCS leakage rate into the secondary containment of 5 gpm. Two percent of the iodine activity in the leakage is assumed to flash and become airborne. This fraction is based on the facts that ECCS temperatures remain below 200 degrees Fahrenheit for the duration of the accident, and because ECCS fluid pH is maintained above 7.0, thereby minimizing iodine vaporization potential. This leak rate is assumed to begin at the onset of the accident and to continue throughout the 30-day duration of the postulated accident.

Regulatory Guide 1.183 accident isotopic release specification allows deposition of iodine in the suppression pool. Essentially all of the iodine is assumed to remain in solution as long as the pool pH is maintained at or above a level of 7.0. Fermi 2 procedures will be reviewed and revised as necessary to include directions for operators to manually initiate the SLC System upon detection of symptoms indicating core damage (e.g., primary containment high radiation). The analysis includes the following assumptions:

1. Borated solution injection is completed within 6 hours following the accident,

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2. Chlorine-bearing materials present on exposed cables inside containment, potentially subject to radiolytic breakdown and carryover of the free chlorine radicals as hydrochloric acid to the suppression pool, can be conservatively represented as approximately 5,792,250 square centimeters of Hypalon with a 0.514 centimeter thickness (80% of the average cable radius), and
3. A minimum mass of 1990 pounds of sodium pentaborate, corresponding to the TS Figure 3.1.7-1 minimum net tank volume of 2712 gallons and minimum concentration by weight of 8.5%, is delivered into the suppression pool.

The calculation results demonstrate that the buffering effect of the boron solution maintains the suppression pool pH above 7.0 for the 30-day duration of the postulated LOCA.

LOCA Transport Inputs

At the beginning of the event, the reactor building exhaust fans are tripped. The reactor building (i.e., secondary containment) is then exhausted by the SGT System continuing the building's negative pressure, thus precluding unfiltered exhaust except for the period of the initial drawdown.

If the main steam lines and the main condenser were to remain intact, the MSIV leakage would eventually collect in the main condenser. However, the analysis assumes that only the portion of the main steam lines between the reactor and the third MSIV in each line remains intact. Although the main steam lines downsteam of the third MSIVs and the condenser itself are expected to remain intact, no credit is taken for these components in the analysis. The shortest main steam line volume, between the outboard and the third MSIVs, and the proposed limit on a single steam line leakage (100 SCFH), are used to determine that an initial 8 hours of holdup inside the main steam lines may be credited. The results presented reflect credit for this delay.

The analysis assumes that unfiltered accident activity is released from the turbine building, without credit for any deposition holdup or dilution in the turbine building.

LOCA Removal Inputs

The activity of elemental iodine and aerosols released from the core into the drywell is assumed to be reduced by deposition (i.e., plate-out) and settling in the drywell utilizing the Powers natural deposition model in the RADTRAD code. Containment leakage into the reactor building is collected by the SGT System which exhausts the reactor building, via filters, and reduces releases. The deposition removal mechanisms are characteristics of the AST methodology and represent a change in the plant design and licensing basis.

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Main steam line pipe deposition was modeled using the Brockmann-Bixler model contained in the RADTRAD computer code, as applied to horizontal segments only.

No credit is taken for holdup or plate-out in the main steam lines beyond the third MSIV. Additionally, no credit is taken for holdup and plate-out in the main condenser. Main steam line deposition was based on using the highest flows in the shortest steam lines (i.e., most rapid transport, and least deposition).

A filter efficiency of 99 percent for iodine was used in the analysis for the SGT System, and a filter efficiency of 99.75 percent for iodine for the CREF system intake (representing 95% efficiency for the makeup filter and 95% efficiency for the recirculation filter), consistent with the current Fermi 2 UFSAR.

3.2 FHA Inputs and Assumptions

The Fermi 2 licensing basis describes two scenarios for the Fuel Handling Accident:

1) FHA involving recently irradiated fuel assumed to occur 24 hours post-shutdown, and credits operability of the SGT System and Secondary Containment isolation and filtration systems
2) FHA involving fuel that has decayed long enough post-shutdown and is no longer recently irradiated, and takes no credit for any ESF mitigation system

Currently, the fuel used in the Fermi 2 reactor is the General Electric (GE) 9X9 fuel rod assembly type (GE11); however, it is planned that future core designs may include the 10X10 GE fuel rod assembly (GE14). The radiological consequences associated with an FHA involving the GE14 fuel type are more limiting than those for the GE11 fuel; therefore, the GE14 was previously used in the FHA AST analysis as a bounding case. However, for the current 9th and the upcoming 10th fuel cycles, the GE11 is the only fuel planned to be used; therefore, the FHA analyzed in this submittal includes both types of fuel.

Furthermore, it has been determined that the GE14 fuel type is expected to meet the limits in Footnote 11 of RG 1.183; however, the GE11 fuel has the potential to exceed these limits. Hence, the current Fermi 2 licensing basis includes provisions, previously approved in Reference 5, to address the potential for the GE11 fuel exceeding the burnup limits in Footnote 11 of the RG. This submittal has no effect on these previously approved provisions; however, the associated analyses have been updated using the revised core inventory and off-site χ/Q's consistent with those used in the evaluation of the DBA LOCA in support of this submittal.

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The key inputs used in the FHA analysis are included in Tables 7 and 8 and are consistent with those used in the analysis approved by the NRC in Reference 5. The re-analyses of the consequences of the FHA are based on the same fuel damage assumptions, but include modified core source terms and off-site χ/Q's consistent with those described herein for the LOCA analysis.

The FHA scenario for recently irradiated fuel assumes the SGT System initiates following an initial 7.2-second period of unfiltered reactor building ventilation release. Control room dose is integrated over 30 days following the accident.

The analysis assumes a conservative bounding normal unfiltered outside air makeup and that the CREF System and control room isolation are not initiated.

4.0 Results

4.1 LOCA Results

The radiological consequences of the DBA LOCA were evaluated using the RADTRAD computer code, using the inputs and assumptions discussed in Section 3.1. The post-accident doses are the result of four distinct activity releases as discussed below.

<u>Primary to Secondary Containment Leakage</u>

Prior to completing secondary containment draw down, this leakage is released to the outside without mixing. After the draw down period, this leakage is captured in the secondary containment (reactor building) and transferred directly to the SGT System without mixing, and is then filtered and released to the environment through the SGT System exhaust stack.

<u>Primary Leakage, Secondary Containment Bypass</u>

This pathway considers piping systems from primary containment to points outside of secondary containment and then to the environment. Five percent (not including the MSIV leakage) of the allowed primary containment leakage is analyzed as bypassing the secondary containment and being released unfiltered by this pathway.

<u>ECCS Leakage into the Secondary Containment</u>

This leakage of 5 GPM is assumed to start immediately after the onset of a LOCA and continue for 30 days. It is filtered by the SGT System prior to release to the environment.

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MSIV Leakage from the Primary Containment into the Environment or Turbine Building

MSIV leakage analysis is assumed to no longer be controlled by the MSIV Leakage Control System, so the MSIV leakage is considered as an unfiltered ground level release.

The postulated exposure to the control room occupants includes immersion and inhalation contribution resulting from inleakage into the control room from the primary containment (direct and bypass leakage), leakage from ECCS, and MSIV leakage releases.

The dose from the following external sources was also considered and was shown to be very small due to the significant structural shielding provided.

- External cloud dose from the primary containment, secondary containment bypass, ECCS, and MSIV leakage releases.
- Direct dose from the primary containment contained accident activity.
- Direct dose from the secondary containment contained accident activity.
- Direct shine from the SGT System filters.

The LOCA control room dose is based on an assumed unfiltered inleakage rate of 900 CFM to the control center. Table 9 presents the results of the LOCA radiological consequence analysis. As indicated, the control room, EAB, and LPZ calculated doses are within the regulatory limits for implementation of AST.

4.2 FHA Results

The radiological consequences of the FHA were analyzed using the inputs and assumptions discussed in Section 3.2. The doses resulting from a postulated FHA are given in Tables 10a and 10b. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits for AST implementation for all analyzed scenarios and fuel types. The analysis indicates that GE11 and GE14 fuel types meeting Regulatory Guide 1.183, Footnote 11 limits become non-recently irradiated 107 and 151 hours, respectively, after no longer occupying part of a critical reactor core.

For the FHA scenario occurring 24 hours after shutdown with SGT System in operation and fuel meeting Regulatory Guide 1.183, Footnote 11 limits, the calculated doses for the control room, EAB, and LPZ are within regulatory limits for AST implementation.

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For GE11 fuel not meeting the limits in Footnote 11 of RG 1.183, the current licensing basis analysis has been re-evaluated using core source term and χ/Q's consistent with other DBAs. The analysis indicated that the GE11 fuel type becomes non-recently irradiated 37 days after no longer occupying part of a critical reactor core. The results, shown in Table 10c, demonstrate that the calculated doses for the control room, EAB, and LPZ are within the associated regulatory limits.

5.0 Atmospheric Relative Concentrations

The χ/Q values for each of the release-intake combinations are summarized in Tables 11 and 12. Table 11 lists χ/Q values used for the control room dose assessments. These were calculated by the ARCON96 computer code using site-specific hourly meteorological data in a five-year period of record. Table 12 list χ/Q values for the EAB and LPZ boundaries. These were calculated using the PAVAN code and Regulatory Guide 1.145 guidance using the same five-year record of site hourly meteorological data.

6.0 Post-Accident Suppression Pool Water Chemistry Management

The re-evolution of iodine from the suppression pool is strongly dependent on pool pH. The analysis assumed that the borated solution of the SLC System was injected within 6 hours of the onset of a DBA LOCA and mixed within the suppression pool. The modeling of the containment cabling conservatively maximized the production of hydrochloric acid. The analysis demonstrated that the suppression pool pH 30 days post-LOCA is greater than 7. The final pH and other related parameters are presented in Table 13.

7.0 Summary and Conclusions

As shown in Tables 9 and 10, the DBA LOCA and FHA radiological consequence analyses demonstrate that the post-accident offsite and control room doses remain within regulatory limits following AST implementation. Furthermore, it was determined that continued compliance with NUREG-0737, Item II.B.2, will be maintained.

Implementation of the AST provides the basis for several changes to the licensing and design bases for Fermi 2. The principal changes affect primary containment and MSIV allowable leakage, Control Center allowable inleakage, and elimination of requirements for the MSIV Leakage Control System. Additionally, a new pH control function is identified for the Standby Liquid Control System.

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

8.0 Other Unaffected Design Basis Accidents

Regulatory Guide (RG) 1.183 states that implementation of an AST and any associated facility modification should be supported by evaluations of all significant radiological and non-radiological impacts of the proposed actions. Furthermore, the RG states that, as a minimum, the DBA LOCA must be re-analyzed using the guidance in Appendix A of the RG.

The Fermi 2 UFSAR includes discussion on several other DBAs that could potentially result in radiological releases outside the primary containment. The UFSAR description includes evaluation of offsite dose consequences at the Exclusion Area Boundary (EAB) and the Low population Zone (LPZ). The discussion below evaluates the impact of implementing the AST on these DBAs and concludes that the analyses in the UFSAR are not affected by the changes proposed in this license amendment.

8.1 Main Steam Line Break

The Main Steam Line Break (MSLB) accident is described in Section 15.6.4 of the UFSAR. The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blow down rates. High steam line flow signal initiates closure of the MSIVs and reactor scram. The break mass released includes that amount in the steam line and connecting lines at the time of the break, plus the amount that passes through the MSIV prior to closure. There is no fuel damage as a result of this accident. The only activity available for release from the break is that present in the reactor coolant and steam lines prior to the break.

The transport pathway is a direct unfiltered release to the environment. The MSIV detection and closure time of 10.5 seconds results in a discharge of approximately 112,000 pounds of steam from the break. All the activity in this discharge is assumed to become airborne. Two activity release cases corresponding to the pre-accident spike and maximum equilibrium concentration allowed by TS of 4.0 μCi/gm and 0.2 μCi/gm dose equivalent I-131, respectively, were assumed. For both cases the resulting calculated dose at the EAB and LPZ boundaries, as reported in the UFSAR, are well within the regulatory limits.

The changes proposed in this license amendment include an increase in the allowed secondary containment bypass leakage and MSIV allowable leakage, deletion of the MSIV LCS, and increase in the secondary containment draw down time and in the unfiltered leakage into the control room envelope. These changes and the implementation of the AST have no impact on the MSLB accident analysis as described in the UFSAR.

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8.2 Control Rod Drop Accident

The Control Rod Drop Accident (CRDA) is described in Section 15.4.9 of the UFSAR. The CRDA involves the rapid removal of the highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. Although analysis shows that the energy deposition that results from this event is inadequate to damage fuel pellets or cladding, for dose consequence analysis, the UFSAR assumes 1.85 percent of the fuel rods in the full core being damaged, and melting occurring in 0.77 percent of the damaged rods (i.e., 0.014% of the core). A core average radial peaking factor of 1.50 is used in the analysis. For releases from the breached fuel, 10% of the core inventory of noble gases and iodines are assumed to be in the fuel gap. For releases attributed to fuel melting, 100% of the noble gases and 50% of the iodines are assumed to be released to the reactor coolant.

The transport pathway consists of carryover with steam to the turbine condenser prior to MSIV closure and leakage from the condenser to the environment. No credit is taken for the turbine building. Of the activity released from the fuel, 100 percent of the noble gases and 10 percent of the iodines are assumed to be carried to the condenser before MSIV closure is complete. Of the activity reaching the condenser, 100 percent of the noble gases and 10 percent of the iodines (due to partitioning and plate-out) remain airborne. The activity airborne in the condenser is assumed to leak directly to the environment at a rate of 1.0 percent per day. Radioactive decay is accounted for during residence in the condenser; it is neglected, however, after release to the environment.

The calculated exposures from the design basis analysis as reported in the UFSAR are well within the guidelines of 10 CFR 100.

The changes proposed in this license amendment include an increase in the allowed secondary containment bypass leakage and MSIV allowable leakage, deletion of the MSIV LCS, and increase in the secondary containment draw down time and in the unfiltered leakage into the control room envelope. These changes and the implementation of the AST have no impact on the CRDA analysis as described in the UFSAR.

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9.0 Impact on Previous Submittals

On September 26, 2002, Detroit Edison submitted a proposed license amendment (Letter NRC-02-0072) to revise the requirements of surveillance SR 3.7.3.6 associated with the verification of CREF system ducting unfiltered inleakage. The amendment proposed crediting the performance of an integrated Tracer Gas test of the Control Room Envelope (CRE) while in the recirculation mode to satisfy the requirements of SR 3.7.3.6. The proposed amendment stated that AST calculations prepared in support of a full scope AST submittal demonstrate that regulatory dose acceptance criteria can be met with significantly higher control room unfiltered inleakage than the 12 CFM assumed in the current design basis. In a subsequent conference call, the NRC informed Detroit Edison that compensatory measures would be required if the measured inleakage by Tracer Gas testing exceeded the 12 CFM assumption in the current design basis. Detroit Edison plans to revise the license amendment request submitted in September 2002 to address NRC comments. It is anticipated that NRC approval of this AST submittal would result in updating the current design basis CRE unfiltered inleakage such that compensatory measures would no longer be required.

10.0 Control Room Habitability

This submittal is not intended to provide a response to the Control Room Habitability (CRH) concerns for which NRC guidance is still being finalized and has not been published in its final version. However, the AST analysis provides an increased margin for control room unfiltered inleakage. The analysis supports revising the design basis CRE unfiltered inleakage from 12 to 900 CFM; hence, this revision will facilitate addressing the CRH concerns, as necessary, in future submittals.

Table 1: Core Inventory

- Based on GE 9x9 bundle
- Core Average Exposure: 35 GWD/MTU (unless noted otherwise)
- 42% Void Fraction (average axial)
- Bundle Average Power = 4.58 MW
- Bundle Average Enrichment = 4 % weight ^{235}U
- 0.172 kg Uranium/bundle

Notes: † Values taken from RADTRAD BWR_DEF.NIF
†† Values conservatively based on 60 GWD/MTU exposure

Isotope	Activity Ci/MW		Isotope	Activity Ci/MW
Co-58†	1.5290E+02		Te-131m	5.2460E+03
Co-60†	1.8300E+02		Te-132	3.8230E+04
Kr-85	3.7360E+02		I-131	2.6570E+04
Kr-85m	6.6930E+03		I-132	3.9010E+04
Kr-87	1.3430E+04		I-133	5.5000E+04
Kr-88	1.8630E+04		I-134	6.0780E+04
Rb-86	4.7670E+01		I-135	5.2350E+04
Sr-89	2.6090E+04		Xe-133	5.4120E+04
Sr-90	3.2950E+03		Xe-135	1.4510E+04
Sr-91	3.2630E+04		Cs-134	4.7930E+03
Sr-92	3.4630E+04		Cs-136	1.4630E+03
Y-90	3.4050E+03		Cs-137	4.2700E+03
Y-91	3.3870E+04		Ba-139	4.8430E+04
Y-92	3.4970E+04		Ba-140	4.8770E+04
Y-93	2.6560E+04		La-140	5.0790E+04
Zr-95	4.5750E+04		La-141	4.4220E+04
Zr-97	4.3220E+04		La-142	4.3200E+04
Nb-95	4.6090E+04		Ce-141	4.4770E+04
Mo-99	4.9880E+04		Ce-143	4.1420E+04
Tc-99m	4.4280E+04		Ce-144	3.7900E+04
Ru-103	4.1830E+04		Pr-143	4.0410E+04
Ru-105	2.8260E+04		Nd-147	1.8000E+04
Ru-106	1.5580E+04		Np-239††	6.6340E+05
Rh-105	2.6240E+04		Pu-238††	2.5610E+02
Sb-127	2.2780E+03		Pu-239††	9.6780E+00
Sb-129	8.5070E+03		Pu-240††	2.6620E+01
Te-127	2.2440E+03		Pu-241††	5.1260E+03
Te-127m	3.7990E+02		Am-241††	6.8800E+00
Te-129	8.0840E+03		Cm-242††	3.1290E+03
Te-129m	1.6390E+03		Cm-244††	6.0240E+02

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Table 2: Personnel Dose Inputs

Input/Assumption	Value
Onsite Breathing Rate	3.47E-04 m^3/sec
Offsite Breathing Rate	0-8 hours: 3.47E-04 m^3/sec 8-24 hours: 1.75E-04 m^3/sec 1-30 days: 2.32E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

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Table 3: Key Analysis Inputs and Assumptions	
Release Inputs - LOCA Radionuclide Alternative Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-S Based Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3499 MWt
Fission Product Release Fractions for LOCA	**RG 1.183, Table 1** **BWR Core Inventory Fraction Released Into Containment** Group / Gap Release Phase / Early In-vessel Phase / Total Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002
Fission Product Release Timing (Per RG 1.183, the release phases are modeled sequentially)	**RG 1.183, Table 4** **LOCA Release Phases for BWRs** Phase / Onset / Duration Gap Release 2 min 0.5 hr Early In-Vessel 0.5 hr 1.5 hr

RG 1.183, Table 1 — BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

RG 1.183, Table 4 — LOCA Release Phases for BWRs

Phase	Onset	Duration
Gap Release	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.5 hr

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Table 4: Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Drywell Free Volume	163,730 cubic feet
Suppression Chamber Air Space Volume	130,900 cubic feet
Primary Containment Leak Rate	0.5% per day for first 24 hours 0.25% per day thereafter
Total MSIV leak rate	250 scfh (100 scfh assumed for each of the two shortest lines) 125 scfh after 24 hours
Reactor Building Net Free Volume (Mixing/Holdup is Not Credited)	2.80E+06 cubic feet
SGT System Initiation Time (on drywell pressure)	SGTS is assumed to initiate on high drywell pressure at time zero. Credit for SGTS filtration is assumed following the completion of the Secondary Containment drawdown.
Secondary Containment Draw down Time	15 minutes from start of gap release. (approximately 17 minutes following the initiating event)
Secondary Containment Bypass[1]	5.0% of primary containment leak rate
ECCS Systems Leakage into Secondary Containment: Quantity: Initiation Time: Leak Rate: Flashing Fraction:	 949,200 gallons 0 5 GPM 2%
<u>Release Location</u> ECCS/Containment Leakage MSIV Leakage	 SGTS Stack TB HVAC Exhaust

1 Analyzed with no piping deposition credit. During leak rate surveillance, leak rates for secondary containment bypass leakage paths may be adjusted to account for piping deposition credit evaluated using the RADTRAD Brockmann-Bixler model.

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Table 5: Key LOCA Analysis Inputs and Assumptions	
Transport Inputs – Control Room Parameters	
Input/Assumption	**Value**
CREF System Initiation (on drywell pressure)	CREF is assumed to initiate on high drywell pressure at time zero
Control Room Total Volume (per UFSAR Table 15.6.5-1)	252,731 cubic feet
Control Room "Shine" Volume (per UFSAR Table 15.6.5-1)	56,960 cubic feet
CREF System Air Intake Flow Rate (1800 CFM proportioned to the "shine" volume)	405.7 CFM
Control Room Filtered Recirculation Rate (1200 CFM proportioned to the "shine" volume)	270.5 CFM
Control Room Unfiltered Inleakage Rate (900 CFM proportioned to the "shine" volume)	202.8 CFM

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Table 6: Key LOCA Analysis Inputs and Assumptions	
Removal Inputs	
Input/Assumption	**Value**
Containment Spray Removal Rates	Not Credited
Aerosol Natural Deposition Coefficients Used in the Drywell	Credit is taken for natural deposition of aerosols based on equations for the Power's model in NUREG/CR-6189 and input directly into RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.
Main Steam Lines Deposition/Plate-out	Calculated for horizontal segments only using RADTRAD Brockmann-Bixler model.
Main Steam Line and Condenser Holdup Credit for MSIV Leakage	No credit is taken for holdup and plate-out downstream of the third MSIVs or in the condenser since these components have not been evaluated for seismic ruggedness.
SGTS Filter Iodine Efficiency	99%
CREF Intake Filter Efficiency CREF Recirculation Filter Efficiency	99.75% 95%

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Table 7: Key Analysis Inputs and Assumptions	
Release Inputs - FHA Radionuclide Alternative Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-S Based Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3499 MWt
Fission Product Gap Release Fractions for FHA	**RG 1.183, Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** **<u>Group</u>** **<u>Fraction</u>** I-131 0.08 Kr-85 0.10 Other Noble Gases 0.05 Other Halogens 0.05 Alkali Metals 0.12

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Table 8: Key FHA Analysis Inputs and Assumptions

Input/Assumption	**Value**
Core Damage	172 rods (GE14 fuel) 140 rods (GE11 fuel)
Radial Peaking Factor	1.7
Fuel Decay Period	GE14: 151 hours (SGT System not operating) GE11: 107 hours (SGT System not operating) 24 hours (SGT System operating)
Fuel Pool Water Iodine Decontamination Factor	DF = 200
Release Period	Less than 2 hours based on a Refuel Floor Volume of 950,000 ft^3 and an assumed ventilation rate of 95,000 cfm
Release Location	• Reactor building vent stack (SGT System not operating) • SGT System Stack (SGT System operating)
Control Room Fresh Air Makeup Rate	4000 cfm
CREF System Initiation	No Credit Taken

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Table 9: LOCA Radiological Consequence Analysis

Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	3.93	5
EAB	Maximum, 2 hours	4.38	25
LPZ	30 days	1.72	25

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Table 10a: FHA Radiological Consequence Analysis (for GE14 [10X10] Fuel Meeting Regulatory Guide 1.183 Footnote 11 Limits)

Location	Duration	TEDE (rem) No SGTS (at 151 hours)	TEDE (rem) With SGTS (at 24 hours)	Regulatory Limit TEDE (rem)
Control Room	30 days	4.70	0.31	5
EAB	Maximum 2 hours	0.27	0.17	6.3
LPZ	30 days	0.06	0.04	6.3

Table 10b: FHA Radiological Consequence Analysis (for GE11 [9x9] Fuel Meeting Regulatory Guide 1.183 Footnote 11 Limits)

Location	Duration	TEDE (rem) No SGTS (at 107 hours)	TEDE (rem) With SGTS (at 24 hours)	Regulatory Limit TEDE (rem)
Control Room	30 days	4.70	0.26	5
EAB	Maximum 2 hours	0.27	0.15	6.3
LPZ	30 days	0.07	0.04	6.3

Table 10c: FHA Radiological Consequence Analysis (for GE11 [9x9] Fuel Not Meeting Regulatory Guide 1.183 Footnote 11 Limits)

Location	Duration	No SGTS (at 37 days)		With SGTS (at 24 hours)		Regulatory Limits	
		Whole body	Thyroid	Whole body	Thyroid	Whole body	Thyroid
Control Room	30 days	0.02	28	0.25	14.2	5	30
EAB	Max 2 hours	0.01	1.4	0.28	0.78	6.25	75
LPZ	30 days	0.003	0.32	0.07	0.18	6.25	75

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Table 11:
Control Room χ/Q (sec/m^3) Values[1,2]

Time Period	LOCA SGTS[3]	LOCA MSIV[3]	FHA No SGTS	FHA w/ SGTS
0 – 7.2 sec (FHA at 24-hrs)[4]				4.03E-3
0 – 2 hrs	6.18E-4	3.10E-4	4.25E-3	3.65E-3
2 - 8 hrs	4.53E-4	2.33E-4		
8 - 24 hrs	1.88E-4	9.93E-5		
1 - 4 days	1.26E-4	7.08E-5		
4-30 days	8.70E-5	5.48E-5		

1 The values above were obtained using ARCON96. Elevated (stack) releases were evaluated as zero-velocity vent releases, otherwise ground calculations were performed.

2 Control room intake χ/Q values are applicable for control room inleakage.

3 Including a factor of four reduction (per Reference 18) for control room dual inlet designs with manual selection control and redundant radiation detectors within each air inlet. CR intakes are separated by more than 45 degrees for each release point. Additionally, the SGTS and MSIV leakage release points are oriented such that they cannot simultaneously impact both control room intakes.

4 After this initial period, the 0 – 2 hour value applies.

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Table 12:
EAB and LPZ χ/Q (sec/m^3) Values

Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 - 2 hours	2.09E-4	4.86E-5 (FHA)*
0 - 8 hours	-	2.17E-5 (LOCA)
8 - 24 hours	-	1.45E-5 (LOCA)
1 - 4 days	-	6.02E-6 (LOCA)
4 - 30 days	-	1.71E-6 (LOCA)

* The 0 – 2 hours χ/Q was conservatively used for the FHA LPZ dose calculation over 30 days.

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Table 13: Suppression Pool pH Results (SLC System Sodium Pentaborate Inventory = 1990 lb_m)	
Time	**pH**
Initial Pool pH	5.6
0-6 hours	>7*
30 days	7.5

* Between 0 and 6 hours, the pH analysis credits Cs released from the fuel (i.e. amount consistent with Cs source term release fraction). After 6 hours, all Sodium Pentaborate is assumed to be in the Suppression Pool to maintain the pH above 7.

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

REFERENCES

1. U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962

2. U. S. Nuclear Regulatory Commission Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

3. U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

4. NRC Letter to Detroit Edison, "Fermi 2 – Issuance of Amendment Re: Revision of Fission Product Release Timing for Design-basis Accidents Described in Fermi 2 Updated Final Safety Analysis Report (TAC No. MB0597)," July 12, 2001

5. NRC Letter to Detroit Edison, "Fermi 2 – Issuance of Amendment Re: Reevaluation of Fuel Handling Accident, Selective Implementation of 10 CFR Part 50.67 (TAC No. MB0956)," September 28, 2001

6. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995

7. 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-2/V2/R6, Volume 2, Section F7, September 1998

8. 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, April 1998

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10. T. J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, U. S. Nuclear Regulatory Commission, November 1982

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

11. J. V. Ramsdell and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, U. S. Nuclear Regulatory Commission, May 1997

12. U. S. Nuclear Regulatory Commission Memorandum from J. L. Birmingham to C. A. Carpenter, "Summary of February 1-2, 2000, Meeting with the Nuclear Energy Institute (NEI) Regarding Control Habitability and NEI 99-03," March 23, 2000

13. Nuclear Energy Institute, NEI 99-03, "Control Room Habitability Assessment Guidance," Revision A, May 2001

14. U. S. Nuclear Regulatory Commission Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1, December 1973

15. ANSI/ANS-2.5-1984, "Standard for Determining Meteorological Information at Nuclear Power Sites"

16. U. S. Nuclear Regulatory Commission Safety Guide 23, "Onsite Meteorological Programs," February 17, 1972

17. U. S. Nuclear Regulatory Commission Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2, June 1974

18. U. S. Nuclear Regulatory Commission Standard Review Plan 6.4, "Control Room Habitability Systems," Revision 2, July 1981

19. U. S. Nuclear Regulatory Commission Memorandum, Initial Screening of Candidate Generic Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump," April 30, 2001

NRC-03-0007
ENCLOSURE 2

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR
FULL SCOPE IMPLEMENTATION OF
ALTERNATIVE SOURCE TERM

REGULATORY GUIDE 1.183 COMPLIANCE

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
3.1 Fission Product Inventory 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 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 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. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. 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. 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.	IN COMPLIANCE ORIGEN-S analyses are used to determine core inventory, considering footnote 9. Power level used is 3499 MWt including a two percent uncertainty factor as per footnote 8 (3430 x 1.02 = 3499). All analyses were performed using a core inventory based on a 9X9 (GE11) fuel bundle design. The Boiling Water Reactor Owners Group (BWROG) determined this fuel type to be the bounding fuel bundle design. A maximum core radial peaking factor of 1.7 is used for the FHA event since it does not involve the entire core.

[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.

[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.

REGULATORY GUIDE 1.183 COMPARISON

<table>
<tr><th colspan="2">FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections</th></tr>
<tr><th>Regulatory Guidance Assumption</th><th>Basis of Compliance</th></tr>
<tr><td>3.2 Release Fractions[10]
The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 1
BWR Core Inventory Fraction Released Into Containment
<table>
<tr><th>Group</th><th>Gap Release Phase</th><th>Early In-Vessel Phase</th><th>Total</th></tr>
<tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr>
<tr><td>Halogens</td><td>0.05</td><td>0.25</td><td>0.3</td></tr>
<tr><td>Alkali Metals</td><td>0.05</td><td>0.20</td><td>0.25</td></tr>
<tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr>
<tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr>
<tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr>
<tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr>
<tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr>
</table>
</td><td>IN COMPLIANCE
The fractions from Table 1 are used for the DBA LOCA calculations.</td></tr>
<tr><td>Table 2 (for PWRs - not included).

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. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.</td><td>A maximum core radial peaking factor of 1.7 is used for the FHA</td></tr>
</table>

[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.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
Table 3[11] **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12	Current Fermi 2 licensing basis for FHA includes consideration of the potential for a limited number of fuel rods in one fuel type in future cycles to exceed the limits in Footnote 11. The AST FHA reanalysis described in this submittal has no impact on the current licensing basis provisions associated with fuel exceeding the limits in Footnote 11.
3.3 Timing of Release Phases Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] 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. **Table 4** **LOCA Release Phases** **Phase** — PWRs **Onset** / **Duration** — BWRs **Onset** / **Duration** Gap Release — 0 sec / 0.5 hr — 2 min / 0.5 hr Early In-Vessel — 0.5 hr / 1.3 hr — 0.5 hr / 1.5 hr	IN COMPLIANCE The BWR LOCA phase onsets and durations from Table 4 are used. LOCA is modeled in a linear fashion. FHA is modeled as an instantaneous release.

[11] 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.

[12] In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable for the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	
3.4 Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** <u>Group</u> — <u>Elements</u> Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium — Ce, Pu, Np	IN COMPLIANCE The 60 nuclides identified as being potentially important dose contributors to total effective dose equivalent (TEDE) in the RADTRAD code are used. The nuclides used encompass those listed in RG 1.183, Table 5.
3.5 Chemical Form 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.	IN COMPLIANCE
3.6 Fuel Damage in Non-LOCA DBAs 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.	IN COMPLIANCE For the FHA, the assumed fraction of fuel elements for which the clad is breached is unchanged from that approved in Amendment 144.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
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.	
4.1 Offsite Dose Consequences **4.1.1** The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	IN COMPLIANCE
4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	IN COMPLIANCE Federal Guidance Report 11 dose conversion factors (DCFs) are used.
4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	IN COMPLIANCE RADTRAD default values to three significant figures are utilized.
4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	IN COMPLIANCE Federal Guidance Report 12 dose conversion factors (DCFs) are used.

[13] The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
4.1.5 TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	IN COMPLIANCE
4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	IN COMPLIANCE
4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.	IN COMPLIANCE
4.2 Control Room Dose Consequences **4.2.1** The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	IN COMPLIANCE All significant sources of radiation, including shine, were considered.

[14] With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50 59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	IN COMPLIANCE The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3 The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	IN COMPLIANCE RADTRAD or other conservative methodologies are utilized.
4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," for guidance.	IN COMPLIANCE The Control Room Emergency Filtration (CREF) System is credited in the LOCA analysis. No credit for CREF is taken in the FHA analysis.
4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	IN COMPLIANCE Such credits are not taken.
4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	IN COMPLIANCE RADTRAD default values to three significant figures are utilized.
4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a	IN COMPLIANCE The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.

[15] The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT and RADTRAD incorporate suitable methodologies.

[16] This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	
4.3 Other Dose Consequences The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	IN COMPLIANCE Assessments indicate that doses would be significantly lower than currently reported.
5.1.1 Analysis Quality The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	IN COMPLIANCE
5.1.2 Credit for Engineered Safeguard Features Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	IN COMPLIANCE The analyses take credit for Standby Liquid Control (SLC) System operation. The SLC System is required to be operable by technical specifications, and is powered by emergency power. The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures.
5.1.3 Assignment of Numeric Input Values The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	IN COMPLIANCE

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Main Sections	
Regulatory Guidance Assumption	**Basis of Compliance**
5.1.4 Applicability of Prior Licensing Basis Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	IN COMPLIANCE
5.3 Methodology Assumptions Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19." The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	IN COMPLIANCE ARCON96 is used to determine control room atmospheric dispersion values. PAVAN using Regulatory Guide 1.145 methodology is used to determine EAB and LPZ atmospheric dispersion values.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs). These assumptions supplement the guidance provided in the main body of this guide. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.	IN COMPLIANCE A large break LOCA is the assumed design basis case for this evaluation of the performance of containment, release mitigation systems, and the acceptability of siting based on calculated EAB and LPZ doses.
SOURCE TERM ASSUMPTIONS 1. Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	IN COMPLIANCE *Fission Product Inventory*: Core source terms are developed using the ORIGEN-S based Generic Source Term methodology. Parameters are 4% initial enrichment and 35 GWD/MTU core average burnup (with minor conservative exceptions). *Release Fractions:* Release fractions are per Table 1 of R.G. 1.183, and are implemented by RADTRAD. *Timing of Release Phases:* Release Phases are per Table 4 of R.G. 1.183, and are implemented by RADTRAD. *Radionuclide Composition:* Radionuclide grouping is per Table 5 of R.G. 1.183, as implemented in RADTRAD. *Chemical Form:* Treatment of release chemical form is per R.G. 1.183, Section 3.5.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
2. If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	The post-LOCA suppression pool pH has been evaluated. This evaluation includes consideration of the effects of acids and bases created during the LOCA event, the effects of key fission product releases, and the impact of SLC System injection. Suppression pool pH remains above 7 for at least 30 days.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT **3.1** The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	IN COMPLIANCE The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell and suppression chamber air space. The suppression chamber free air volume is included based on expected steam flow from the drywell to the suppression chamber.
3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments." The latter model is incorporated into the analysis code RADTRAD. The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Credit is taken for natural deposition per the methodology of NUREG/CR-6189, as implemented in RADTRAD. No deterministically assumed initial plateout is credited.
3.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays". This simplified model is incorporated into the analysis code RADTRAD. *{Balance of guidance not repeated, since spray is not credited}*	While drywell and suppression chamber sprays are available at Fermi 2, no credit for these sprays is taken for aerosol removal in the LOCA AST reanalysis.
3.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	N/A. No in-containment recirculation filter systems exist at Fermi 2.
3.5 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. Analyses have been performed that determined that the suppression pool

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool. Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	liquid pH is maintained greater than 7; therefore, iodine re-evolution is not expected.
3.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP.	N/A. Fermi 2 does not have ice condensers.
3.7 The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Equilibrium suppression pool temperatures, and resulting containment pressures, including steam partial pressures will be sufficiently low after 24 hours such that a 50% of technical specification leak rates can be conservatively assumed for the 24 to 720 hour period.
For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	N/A, Fermi containment is a Mark I.
3.8 If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	The Fermi 2 primary containment is not routinely purged during power operation. Operation of the purge system is performed for inerting following reactor startup and for de-inerting prior to reactor shutdown, if necessary. Therefore, the analysis does not include any contribution from releases via the purge system prior to containment isolation.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
ASSUMPTIONS ON DUAL CONTAINMENTS **4.** For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.	IN COMPLIANCE
4.1 Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Secondary Containment filtered release credit is taken at 15 minutes after the start of gap release. Gap release begins at ~ 2 minutes after LOCA initiation. Therefore, a period of about 17 minutes is available for achieving a negative pressure of 1/4" W.G. For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions.
4.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions.
4.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	The potential for high wind speeds, impacting the ability of secondary containment to maintain negative pressures for wind speeds not exceeded 95% of the time, is already addressed in UFSAR Subsection 6.2.1.2.2.3, where it is demonstrated that positive pressures would not be experienced.
4.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that	No credit is taken for dilution in secondary containment.

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FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
impede stream flow between the release and the exhaust. **4.5** Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Bypass leakage has been analyzed at 5% of L_A in support of a license change of the current limit of 4% of L_A. The baseline dose assessment for this pathway does not credit piping deposition in any of the associated lines. Release of MSIV leakage at Fermi 2 has previously been prevented using the MSIV LCS. This system is no longer credited. MSIV leakage has a separate technical specification limit of 250 scfh total leakage with not more that 100 scfh per line. The dose consequences for releases through this pathway (with piping deposition credit) are separately calculated. Therefore, this leakage is not considered part of the L_A leakage limit, which also has a separate dose consequence assessment. Therefore, while listed as a bypass pathway in UFSAR Table 6.2-2, MSIV leakage need not be considered in the 5% L_A secondary containment bypass allowance. Feedwater piping deposition has been evaluated to determine its effectiveness relative to SGT System filtration, if 95% of L_A were to exhaust through a single penetration. It has also been evaluated with a 5% L_A bypass leakage with and without piping deposition credit. For any of the EAB, LPZ, and CR dose receptors, the relative effectiveness determinations are proposed to be used to adjust

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
	measured leakage for piping deposition credit, before LLRT Type B & C totaling and comparison with L_A limits. Piping deposition credit is determined using the Brockmann-Bixler routines available in RADTRAD. This methodology may be applied to other bypass lines, provided proper calculations are performed and documented.
4.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Compliance with Regulatory Guide 1.52 is discussed in the UFSAR. The SGT System filters are credited at an efficiency of 99% for all iodine chemical forms.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
ASSUMPTIONS ON ESF SYSTEM LEAKAGE **5.** ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems. The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.	IN COMPLIANCE This pathway is analyzed per guidance.
5.1 With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core.
5.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	The design basis 5 GPM leak rate is more than 2 times the acceptance criteria for the sum of the simultaneous leakage from all components in the ESF recirculation systems as addressed in the Program committed to in T.S. 5.5.2 "Primary Coolant Sources Outside Containment", which is Fermi 2's license commitment to item III.D.1.1 of NUREG-0737. Since certain ECCS systems take suction immediately from the suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric tanks are credible only for lines connecting ECCS pump discharges to such tanks, because of relative elevations. The sole

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Regulatory Guidance Assumption	**Basis of Compliance**
	leakage path to a tank vented to atmosphere meeting this condition is the HPCI / RCIC test lines that returns to the CST. These lines are isolated by two normally closed valves, one air operated and one motor operated. The CST contents are demineralized water, which leaked ECCS fluid would quickly turn basic. Therefore, minimal elemental iodine is expected, and as a result, negligible iodine volatilization.
5.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: [Equation not used or repeated here.]	The temperature of leakage does not exceed 212°F.
5.5 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	An airborne release fraction of 2% is used in this accident analysis. Suppression water pH is maintained above 7 for the entire 30 days accident dose assessment period. Under these conditions, virtually none of the iodine will be in elemental form, and organic iodine formation will be inhibited. Because of the subcooled condition no flashing is expected. Aerosols which may spray from a leakage point will

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Regulatory Guidance Assumption	**Basis of Compliance**
	tend to be large and readily settleable. Post-LOCA Reactor Building air change rates are only on the order of 1/day and minimal entrainment is expected. Leaked liquids may collect and evaporate, and if so, pH will increase. Elemental iodine is expected to react with surfaces or other solution components, and a small fraction might vaporize. To provide a conservative allowance for elemental iodine vaporization or organic iodine formation, 2% of the leaked iodines are assumed to become airborne.
5.6 The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.	R.G. 1.52 compliance for SGT and CREF System filters is discussed in the UFSAR. SGT and CREF System filter efficiencies that are used in this accident analysis are the same for all chemical species.

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FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"

Regulatory Guidance Assumption	Basis of Compliance
ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS **6.** For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.	IN COMPLIANCE Because the Fermi 2 MSIV LCS will not be credited in this accident analysis, MSIV leakage will be considered a radioactivity release pathway, and the radiological consequences of such a release are analyzed.
6.1 For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell and suppression chamber air space. The suppression chamber free air volume is included based on expected steam flow from the drywell to the suppression chamber.
6.2 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	MSIV leakage assumed in this accident analysis is 250 scfh for all steam lines and 100 scfh for any one line. A reduction in leakage of 50% is assumed at 24 hours, based on expected containment pressures at that time.
6.3 Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Modeling is per RADTRAD Brockmann-Bixler approach and is based on a well mixed containment model.
6.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Since MSIV LCS is no longer credited, no ESFs are assumed to be available to collect or treat MSIV leakage. Releases are assumed to be from the turbine building exhaust, without credit for holdup or dilution in the condenser or turbine building.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix A, "Assumptions for Evaluating The Radiological Consequences of a LWR Loss-Of-Coolant Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
6.5 A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Main steam piping downstream of the primary MSIVs is credited for piping that is capable of performing their safety function during and following an SSE. No credit is taken for holdup and deposition in piping downstream of the third MSIVs or in the condenser. The modeling is per the RADTRAD Brockmann-Bixler approach.
ASSUMPTION ON CONTAINMENT PURGING **7.** The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 and Generic Letter 99-02.	Containment purging as a combustible gas or pressure control measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at Fermi 2.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
1. SOURCE TERM Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	
1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	IN COMPLIANCE .
1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	IN COMPLIANCE Cesium and Rubidium not included because they are assumed to be particulates with infinite DF (see Section 3. "NOBLE GASES" below).
1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	IN COMPLIANCE
2. WATER DEPTH If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.	IN COMPLIANCE An overall DF of 200 is used.
3. NOBLE GASES The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	IN COMPLIANCE

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
4. FUEL HANDLING ACCIDENTS WITHIN THE FUEL BUILDING **4.1** The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	IN COMPLIANCE Release from the refuel floor to the environment is assumed to be within 2 hours.
4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system[1] should be determined and accounted for in the radioactivity release analyses.	IN COMPLIANCE Credited ESF filter systems conform to R.G. 1.52 as discussed in the UFSAR. A conservative delay of 7.2 seconds is assumed for SGT System filter actuation, when filters are credited.
4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	IN COMPLIANCE Two-hour release to the environment is assumed.
5. FUEL HANDLING ACCIDENTS WITHIN CONTAINMENT For fuel handling accidents postulated to occur within containment:	
5.1 If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable.

[1] These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

[2] Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in technical specifications.

REGULATORY GUIDE 1.183 COMPARISON

FERMI 2 Compliance Matrix for Regulatory Guide 1.183, Rev. 0, Appendix B "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident"	
Regulatory Guidance Assumption	**Basis of Compliance**
5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Not Applicable.
5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Not Applicable.
5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]	Not Applicable.
5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable.

[3] The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

NRC-03-0007
ENCLOSURE 3

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR FULL SCOPE IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION

In accordance with 10 CFR 50.92, Detroit Edison has made a determination that the proposed amendment involves no significant hazards consideration. The proposed amendment requests revisions to the Technical Specifications to reflect the application of alternative source term (AST) assumptions.

AST analyses have been performed based on the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," dated July, 2000.

The AST analyses have been performed with assumed increase in allowable leakage from secondary containment bypass and Main Steam Isolation Valves (MSIVs), and increase in unfiltered inleakage into the Control Room. The analyses take no credit for the operation of the MSIV Leakage Control System or any holdup and plate-out in the main steam system piping beyond the third MSIVs. Licensing basis revisions are proposed to reflect these positions.

1. **The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The implementation of AST assumptions has been evaluated in a revision to the analysis of the Loss of Coolant Accident (LOCA) and an update to the analysis of the Fuel Handling Accident (FHA).

   Based upon the results of the analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting Design Basis Accidents (DBAs) are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan (SRP) Section 15.0.1.

   The requirements for MSIV Leakage Control System operability for eliminating MSIV leakage to the environment are being eliminated. This is acceptable because, with the application of AST, this system is no longer credited in mitigating the consequences of a LOCA or any other DBA.

   The proposed changes also increase the limits on maximum allowable leakage from secondary containment bypass and main steam isolation valves, and on unfiltered inleakage into the Control Room. This is acceptable due to the new assumptions used in calculating Control Room and offsite dose following the affected design basis accident using the AST methodology.

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes do not affect the normal design or operation of the facility before the accident; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequence. The radiological consequences of the analyzed DBAs have been evaluated with application of AST assumptions. The results conclude that the radiological consequences remain within applicable regulatory limits. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The application of AST does not affect the design, functional performance or normal operation of the facility. Similarly, it does not affect the design or operation of any component in the facility such that new equipment failure modes are created. Elimination of the MSIV Leakage Control System cannot create a new accident because it is used as a mitigation system to limit MSIV leakage after the accident has occurred. Similarly, the use of Standby Liquid Control System to buffer suppression pool pH to prevent iodine re-evolution is another mitigation function credited after the accident has occurred and; therefore, cannot create a new accident.

As such the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed changes do not involve a significant reduction in a margin of safety.**

This proposed license amendment involves changes from the original source term developed in accordance with Technical Information Document (TID) 14844 to a new AST, as described in Regulatory Guide 1.183. The results of the DBA analyses and the requested Technical Specification changes, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analysis adequately bounds postulated event scenario. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183 and SRP Section 15.0.1.

The margin of safety is that provided by meeting the applicable regulatory limits. The effect of relaxation of these design and Technical Specification requirements has been analyzed and doses resulting from the design basis accidents have been found to remain within the regulatory limits. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits.

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION

Therefore, operation of Fermi 2 in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Based on the above, Detroit Edison has determined that the proposed license amendment does not involve a significant hazards consideration.

NRC-03-0007
ENCLOSURE 4

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR
FULL SCOPE IMPLEMENTATION OF
ALTERNATIVE SOURCE TERM

INFORMATION SUPPORTING ENVIRONMENTAL CONSIDERATION

INFORMATION SUPPORTING ENVIRONMENTAL CONSIDERATION

Detroit Edison has evaluated the proposed changes against the criteria for identifying licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." Detroit Edison has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). As such, Detroit Edison has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20," Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement. The amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As discussed in Enclosure 3, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The results of the Loss of Coolant Accident (LOCA) and the Fuel Handling accident (FHA) Design Basis Accident (DBA) analyses demonstrate that the radiological acceptance criteria described in 10 CFR 50.67 for the exclusion area boundary (EAB) and the low population zone (LPZ) are met. The EAB and LPZ total effective dose equivalent (TEDE), with an assumed 900 CFM of Control Room unfiltered inleakage, 5% Secondary Containment bypass leakage, and no Main Steam Isolation Valve Leakage Control System (MSIVLCS) credit, represent a small fraction of the regulatory dose limits.

Adoption of the alternative source term and Technical Specification changes which are based on certain conservative assumptions in the alternative source term analyses will not result in modifications to the plant or changes in its operation which could significantly alter the type or amounts of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The alternative source term DBA LOCA and FHA analyses demonstrate that the radiological acceptance criteria described in 10 CFR 50.67 for the control room is met.

INFORMATION SUPPORTING ENVIRONMENTAL CONSIDERATION

Control room exposure to operators is less than the five rem TEDE over 30 days for all design basis accidents.

The alternative source term does not affect the normal design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequences. The implementation of the alternative source term has been evaluated in revisions to two of the limiting design basis accidents at Fermi 2. Based upon the results of the analyses, it has been demonstrated that with the requested changes, the dose consequences of these limiting DBAs are within the regulatory guidance provided by the NRC for use with alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

NRC-03-0007
ENCLOSURE 5

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR
FULL SCOPE IMPLEMENTATION OF
ALTERNATIVE SOURCE TERM

Attached are marked-up pages of the existing TS indicating the proposed changes (Part 1) and a typed version incorporating the proposed changes (Part 2)

NRC-03-0007
ENCLOSURE 5
PART 1

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED TS MARKED UP PAGES

<u>INCLUDED PAGES:</u>

Table of Contents Page ii
3.6-17
3.6-27 and 3.6-28
3.6-42

TABLE OF CONTENTS

Deleted

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	Verify the combined equivalent leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system is ≤ ~~0.04~~ 0.05 L_a when pressurized to ≥ 56.5 psig.	In accordance with the Primary Containment Leakage Rate Testing Program and Inservice Testing Program
SR 3.6.1.3.12	Verify combined MSIV leakage rate for all four main steam lines is ≤ ~~100 scfh~~ 250 scfh and ≤ 100 scfh for any one steam line when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.13	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. --- Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

Delete entire Section 3.6.1.9

3.6 CONTAINMENT SYSTEMS

3.6.1.9 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

LCO 3.6.1.9 Two MSIV LCS subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSIV LCS subsystem inoperable.	A.1	Restore MSIV LCS subsystem to OPERABLE status.	30 days
B. Two MSIV LCS subsystems inoperable.	B.1	Restore one MSIV LCS subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

Delete entire Section 3.6.1.9

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.9.1	Verify each MSIV LCS valve, testable during operation, cycles through at least one complete cycle of full travel.	31 days
SR 3.6.1.9.2	Verify each MSIV LCS valve, not testable during operation, cycles through at least one complete cycle of full travel.	Prior to entering MODE 2 or 3 from MODE 4 if not performed in the previous 31 days
SR 3.6.1.9.3	Perform a system functional test of each MSIV LCS subsystem.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.2	---------------NOTE------------------ Not required to be met for one railroad bay access door until: a. 4 hours after opening for entry, exit, or testing; and b. 12 hours after opening for new fuel receipt activities provided the other door remains OPERABLE and closed. --------------------------------------- Verify all secondary containment equipment hatches, pressure relief doors and railroad bay access doors are closed and sealed.	31 days
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify steam tunnel blowout panels are closed.	Prior to entering MODE 2 or 3 from MODE 4 if not performed in the previous 31 days
SR 3.6.4.1.5	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ ~~567 seconds~~ 12 minutes.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.6	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 3000 cfm.	18 months on a STAGGERED TEST BASIS

NRC-03-0007
ENCLOSURE 5
PART 2

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED TS REVISED PAGES

<u>INCLUDED PAGES:</u>

Table of Contents Page ii
3.6-17
3.6-27 and 3.6-28
3.6-42

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	Verify the combined equivalent leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system is $\leq 0.05\ L_a$ when pressurized to $\geq$ 56.5 psig.	In accordance with the Primary Containment Leakage Rate Testing Program and Inservice Testing Program
SR 3.6.1.3.12	Verify combined MSIV leakage rate for all four main steam lines is $\leq$ 250 scfh and $\leq$ 100 scfh for any one steam line when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.13	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.2	---------------NOTE--------------- Not required to be met for one railroad bay access door until: a. 4 hours after opening for entry, exit, or testing; and b. 12 hours after opening for new fuel receipt activities provided the other door remains OPERABLE and closed. ------------------------------------ Verify all secondary containment equipment hatches, pressure relief doors and railroad bay access doors are closed and sealed.	31 days
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify steam tunnel blowout panels are closed.	Prior to entering MODE 2 or 3 from MODE 4 if not performed in the previous 31 days
SR 3.6.4.1.5	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 12 minutes.	18 months on a STAGGERED TEST BASIS
SR 3.6.4.1.6	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 3000 cfm.	18 months on a STAGGERED TEST BASIS

NRC-03-0007
ENCLOSURE 6

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED LICENSE AMENDMENT FOR
FULL SCOPE IMPLEMENTATION OF
ALTERNATIVE SOURCE TERM

Attached are marked-up pages of the existing TS Bases indicating the proposed changes (For Information Only)

INCLUDED PAGES:

Table of Contents Page ii
B 2.1.1-4
B 2.1.2-1, B 2.1.2-2 & B 2.1.2-3
B 3.1.7-1, B 3.1.7-2 & B3.1.7-7
B 3.3.6.1-7, B 3.3.6.1-12 & B 3.3.6.1-13
B 3.6.1.3-15, B 3.6.1.3-16, B 3.6.1.3-17 & B 3.6.1.3-18
B 3.6.1.9-1, B 3.6.1.9-2 & B 3.6.1.9-3
B 3.6.4.1-6

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*Deleted.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

50.67 "Accident Source Term" Limits

SAFETY LIMIT VIOLATIONS

Exceeding a reactor core SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria," limits~~ (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A (latest approved revision).
3. 10 CFR ~~100.~~

50.67

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with a low probability of failure and sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR ~~100, "Reactor Site Criteria"~~ 50.67, "Accident Source Term" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

BASES

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel, main steam piping, and reactor coolant system are designed and built in accordance with applicable codes and standards, as detailed in Reference 5. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping and 1500 psig for discharge piping. The most limiting of these allowances is the 110% of the suction piping design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

50.67 "Accident Source Term"

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWB-5000.
4. 10 CFR ~~100.~~ 50.67
5. UFSAR Section 3.2.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

INSERT A

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration equivalent to 720 ppm of natural boron, in the reactor coolant at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature limit in SR 3.1.7.2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with reactor water level at Level 8 and including the water volume in the residual

Insert A

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that Iodine will be retained in the suppression pool water (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

INSERT B

The SLC System satisfies the requirements of 10 CFR 50.36(c)(2)(ii) because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

INSERT B1

Insert B

Following a LOCA involving significant fission product release, offsite doses from the accident will remain within 10 CFR 50.67, "Accident Source Term," limits (Ref. 4) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Ref. 3) as long as suppression pool pH is maintained at or above 7. Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool at or above 7.

Insert B1

The SLC System must also be OPERABLE in MODES 1 and 2 to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 4) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 3).

BASES

REFERENCES

1. 10 CFR 50.62.
2. UFSAR, Section 4.5.2.4.
3. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, Final Report, February 1, 1995.
4. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level – Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level – Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level – Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level – Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits. 50.67

This Function isolates the MSL and MSL drains isolation valves.

1.b. Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function, although not credited in the analysis of the pressure regulator failure (Ref. 2), is a back-up to the maximum steam flow limiter, which is credited by this analysis. For

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Primary Containment Isolation

2.a. Reactor Vessel Water Level — Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level — Low, Level 3 Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA. 50.67

Reactor Vessel Water Level — Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function shares common instrumentation with the RPS.

This Function isolates the drywell sumps and TIP isolation valves.

2.b. Reactor Vessel Water Level — Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level — Low Low, Level 2 Function associated with isolation is implicitly assumed to be isolated post-LOCA. 50.67

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Vessel Water Level – Low Low, Level 2 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level – Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Level 2 Initiation Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates 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.

2.c. Drywell Pressure – High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure – High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

50.67

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.7

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 100 limits. The minimum stroke time ensures that isolation does not result in a pressure spike more rapid than assumed in the transient analyses. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

50.67

SR 3.6.1.3.8

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.5 overlaps this SR to provide complete testing of the safety function. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.9

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that each tested valve restricts flow on a simulated instrument line break. The representative sample consists of an approximately equal number of EFCVs (about 15), from different plant locations and operating environments, such that each EFCV is tested at least once every ten years. The representative sample testing reflects the operability status of all EFCVs in the plant. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event evaluated in Reference 5.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 18 month representative sample test frequency is based on the typical performance of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The nominal ten-year maximum limit is based on performance testing. Any EFCV failure will be evaluated per the Corrective Action and the Maintenance Rule programs to determine if additional testing is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 6).

SR 3.6.1.3.10

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. No squib will remain in service beyond the expiration of its shelf life or its operating life. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.11

equivalent

$\leq 0.05\ L_a$.

This SR ensures that the leakage rate of secondary containment bypass leakage paths is ~~less than the specified leakage rate~~. This provides assurance that the assumptions in the radiological evaluations of Reference 1 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The frequency is required by the Primary

Leak rates for secondary containment bypass leakage paths may be adjusted to account for piping deposition credit using the Brockmann-Bixler option of RADTRAD computer code (Ref. 8) resulting in a reduced equivalent leakage.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Additionally, some secondary containment bypass paths (refer to UFSAR 6.2.1.2.2.3) use non-PCIVs and therefore are not addressed by the testing Frequency of 10 CFR 50, Appendix J, testing. To address the testing for these valves, the Frequency also includes a requirement to be in accordance with the Inservice Testing Program.

Secondary containment bypass leakage is also considered part of L_a.

250 scfh, and ≤ 100 scfh for any one steamline,

SR 3.6.1.3.12

The analyses in References 1 and 4 are based on leakage that is less than the specified leakage rate. Leakage through all four main steam lines must be ≤ 100 scfh when tested at ≥ P_t (25 psig). ~~This ensures that MSIV leakage is properly accounted for to assure safety analysis assumptions, regarding the MSIV-LCS ability to provide a positive pressure seal between MSIVs, remain valid.~~ This leakage test is performed in lieu of 10 CFR 50, Appendix J, Type C test requirements, based on an exemption to 10 CFR 50, Appendix J. As such, this leakage is not combined with the Type B and C leakage rate totals. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

MSIVs have separate leakage limits, and the dose consequence of this leakage path is evaluated separately and added to those calculated from primary containment L_a leakage, including secondary containment bypass leakage.

SR 3.6.1.3.13

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 4 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1 gpm times the number of valves per penetration, not to exceed 3 gpm, when tested at 1.1 P_a (≥ 62.2 psig). Additionally, a combined leakage rate limit of ≤ 5 gpm when tested at 1.1 P_a (≥ 62.2 psig) is applied for all hydrostatically tested PCIVs that penetrate containment. The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by Primary Containment Leakage Rate Testing Program.

This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Table 6.2-2.
3. 10 CFR 50, Appendix J, Option B.
4. UFSAR, Section 6.2.
5. UFSAR, Section 15.6.2.
6. GE BWROG B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998.
7. Technical Requirements Manual, Section TR 3.6.3
8. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," April 1998.

Delete entire
Section B 3.6.1.9

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.9 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

BASES

BACKGROUND

The MSIV LCS supplements the isolation function of the MSIVs by containing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSIV LCS consists of two independent subsystems: MSIV LCS Division I is designed to inject air supplied by the control air system into the piping volume bounded by the four pairs of inboard and outboard MSIVs. The MSIV LCS Division II is designed to inject air supplied by the control air system into piping volume bounded by the four pairs of outboard and a third set of motor operated MSIVs (Division II). While the third set of MSIVs are associated with Division II MSIV LCS, the inboard and outboard MSIVs are not considered part of the MSIV LCS System. Refer to LCO 3.6.1.3, "PCIVs," for MSIV OPERABILITY requirements.

The MSIV LCS is designed to operate at a higher pressure than the primary containment after a postulated LOCA to provide protection against potential leakage of radioactive contaminants to the environment.

The MSIV LCS is manually initiated approximately 20 minutes following a DBA LOCA (Ref. 2).

APPLICABLE SAFETY ANALYSES

The MSIV LCS mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are contained. The operation of the MSIV LCS prevents a release of untreated leakage for this type of event.

The MSIV LCS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One MSIV LCS subsystem can provide the required containment of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSIV LCS subsystems must be OPERABLE.

Delete entire Section B 3.6.1.9

BASES

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSIV LCS OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSIV LCS OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.

ACTIONS

A.1

With one MSIV LCS subsystem inoperable, the inoperable MSIV LCS subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE MSIV LCS subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSIV LCS subsystem and the low probability of a DBA LOCA occurring during this period.

B.1

With two MSIV LCS subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

C.1 and C.2

If the MSIV LCS subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed 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.

Delete Entire Section B3.6.1.9

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1 and SR 3.6.1.9.2

Cycling each of the MSIV LCS System valves through at least one complete cycle of full travel demonstrates that the valves are mechanically capable of functioning upon initiation of MSIV LCS. SR 3.6.1.9.1 applies to MSIV LCS valves that are testable during plant operation. These valves can be cycled without affecting plant operation. SR 3.6.1.9.2 applies to MSIV LCS valves that are not testable during plant operation, which includes the motor-operated MSIVs. These valves could affect plant operation if cycled. The SR 3.6.1.9.2 Frequency only requires this surveillance to be performed while shutdown to eliminate the impact on plant operations that would be caused by cycling these valves during plant operation. The 31 day frequency has been found acceptable based upon engineering judgment and operating experience.

SR 3.6.1.9.3

A system functional test is performed to ensure that the MSIV LCS will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock is correct. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.2.6.
2. UFSAR, Section 6.2.6.2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.4

If the steam tunnel blowout panels are open the integrity of the Secondary Containment is lost. Since the steam tunnel blowout panels are inaccessible during plant operation, this SR is only required to be performed during MODE 4, but only if it has been greater than 31 days since the last verification. This frequency has been shown to be adequate based on operating experience, and in view of other indications of the status of the steam tunnel blowout panels available to the operator.

SR 3.6.4.1.5 and SR 3.6.4.1.6

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.5 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw

12 minutes

down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~567 seconds~~. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.6 demonstrates that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 3000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.

2. UFSAR, Section 15.7.4.