September 28, 2004

Mr. William T. O'Connor, Jr.
Vice President - Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: SELECTIVE IMPLEMENTATION OF ALTERNATIVE RADIOLOGICAL SOURCE TERM METHODOLOGY (TAC NO. MB7794)

Dear Mr. O' Connor:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 160 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the design basis and Technical Specifications (TSs) in response to your application dated February 13, 2003, as supplemented July 8, 2003, December 12, 2003, June 4, 2004, July 30, 2004, and September 16, 2004.

The amendment approves the use of an alternative source term methodology in accordance with Title 10 of the *Code of Federal Regulations,* Part 50, Section 50.67, based on a reevaluation of the design-basis loss-of-coolant and fuel handling accidents. In addition to related design-basis changes, the amendment revises the TSs to (1) permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs), (2) increase the allowable secondary containment bypass leakage, (3) delete the MSIV leakage control system, and (4) increase the allowed secondary containment draw-down time.

A copy of the safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

***/RA/***

David P. Beaulieu, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No. 160 to License No. NPF-43
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

<u>DISTRIBUTION</u>

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| DATE | 09/28/04 | 09/28/04 | 09/14/04 | 09/14/04 | 09/21/04 | 09/28/04 |

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DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 160
License No. NPF-43

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Detroit Edison Company (DECo) dated February 13, 2003, as supplemented July 8, 2003, December 12, 2003, June 4, 2004, July 30, 2004, and September 16, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

    <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 160, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. In addition, by Amendment No. 160, the license is amended to authorize selective use of an alternative source term methodology in accordance with 10 CFR 50.67 based on a reevaluation of the design-basis loss-of-coolant and fuel handling accidents. The licensee shall update the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated February 13, 2003, as supplemented July 8, 2003, December 12, 2003, June 4, 2004, July 30, 2004, and September 16, 2004, and the NRC staff's safety evaluation attached to this amendment. The licensee shall submit the revised description authorized by this amendment with the next update of the UFSAR.

4. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 28, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 160

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
| --- | --- |
| Table of Contents Page ii | Table of Contents Page ii |
| 3.6-17 | 3.6-17 |
| 3.6-27 | 3.6-27 |
| 3.6-28 | 3.6-28 |
| 3.6-42 | 3.6-42 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 160 FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

## 1.0 INTRODUCTION

By application dated February 13, 2003, as supplemented July 8, 2003, December 12, 2003, June 4, 2004, July 30, 2004, and September 16, 2004, the Detroit Edison Company (the licensee) requested changes to the design basis and Technical Specifications (TSs) for Fermi 2. The proposed amendment would approve the selective use of an alternative source term (AST) methodology in accordance with Title 10 of the *Code of Federal Regulations,* Part 50, Section 50.67, based on a reevaluation of the design-basis loss-of-coolant accident (LOCA) and fuel handling accidents (FHAs). In addition to related design-basis changes, the amendment would also revise the TSs to (1) permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs), (2) increase the allowable secondary containment bypass leakage, (3) delete the MSIV leakage control system, and (4) increase the allowed secondary containment draw-down time. The proposed changes are discussed in further detail in Section 3.0 of this safety evaluation (SE).

The supplements dated July 8, 2003, December 12, 2003, June 4, 2004, July 30, 2004, and September 16, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 27, 2003 (68 FR 28847).

## 2.0 REGULATORY EVALUATION

In December 1999, the NRC issued 10 CFR 50.67, "Accident source term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their design-basis accident (DBA) analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." A licensee seeking to use an AST is required by 10 CFR 50.67 to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the application. The licensee's application of February 13, 2003, as supplemented, addresses these requirements in proposing to selectively use the AST described in RG 1.183 as the source term in the evaluation of the radiological consequences of the LOCA and FHAs at Fermi 2. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, “Control room,” as the Fermi 2 licensing basis for the LOCA.

This SE addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and 10 CFR Part 50, Appendix A, GDC-19. Except where the licensee has proposed a suitable alternative, the NRC staff used the regulatory guidance in RG 1.183, and Standard Review Plan 15.0-1, “Radiological Consequence Analyses Using Alternative Source Terms,” in doing this review. The NRC staff also considered relevant information in the Fermi 2 Updated Final Safety Analysis Report (UFSAR) and TSs.

As authorized by License Amendment No. 144, dated September 28, 2001, the Fermi 2 design basis includes two fuel handling accidents (FHAs). The first is an analysis done for General Electric fuel product line 11 (GE11) that is based on Safety Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors,” and the whole-body and thyroid dose criteria of 10 CFR 100.11. The second addresses the remaining fuel and is based on RG 1.183.

The licensee proposed a new safety function for the standby liquid control (SLC) system. This new safety function would utilize a characteristic of the injected sodium pentaborate solution to control suppression pool pH. If a pH level of 7.0 or greater is not achieved within 24 hours and maintained for the duration of the event, elemental radioiodine may evolve from the suppression pool. This new safety function would serve to control fission products released into the containment and therefore falls within the intent of 10 CFR Part 50, Appendix A, GDC-41, "Containment atmosphere cleanup." Although the system was not originally reviewed against GDC-41, the NRC staff reviewed the pH control proposal against this design criterion.

## 3.0 <u>TECHNICAL EVALUATION</u>

### 3.1 <u>Accident Dose Calculations</u>

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were done by the licensee in support of this proposed license amendment. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess these impacts. The NRC staff completed independent calculations to confirm the conservatism of the licensee’s analyses. However, the findings of this SE input are based on the descriptions of the analyses and other supporting information submitted by the licensee. Only docketed information was relied upon in making this safety finding.

In accordance with the guidance in RG 1.183, a licensee is not required to reanalyze all DBAs for the purpose of the application, just those affected by the proposed changes. In keeping with this guidance, the licensee evaluated previously analyzed DBAs to decide which, if any, were affected by the proposed amendment. The licensee considered the following DBAs:

- main steam line break (MSLB)
- control rod drop accident (CRDA)

- LOCAs
- FHAs

3.1.1 MSLB/CRDA

The MSLB and CRDA are postulated to occur outside the secondary containment. The proposed changes related to secondary containment draw down, increase in MSIV allowable leakage, and removal of the MSIV leakage control system have no impact on the previously analyzed MSLB, or CRDA. MSLB releases are based on the reactor coolant system (RCS) specific activity TS, which is unaffected by the AST implementation. The current CRDA analysis releases are based on previous regulatory guidance that is identical to that provided for the AST in RG 1.183. As such, these events need not be considered further.

The licensee reanalyzed the LOCA and FHAs. The licensee updated the FHA analyses to reflect changes in the core inventory and $\chi/Q$ values; no other proposed change affected these analyses. For the reanalyses of the LOCA and FHAs, the licensee determined the TEDE at the exclusion area boundary (EAB) for the worst two hour period and the 0-30 day low population zone (LPZ) TEDE. The licensee also evaluated the potential TEDE to control room personnel from these DBAs. The accident-specific sections that follow describe the postulated accidents, the licensee's evaluation of the impact of the proposed changes, and the NRC staff's evaluation.

3.1.2 Loss-of-Coolant Accident

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment in the unlikely event that the emergency core cooling system (ECCS) is not effective in preventing core damage. A LOCA is a failure of the RCS that results in the loss of reactor coolant that, if not mitigated, could result in fuel damage including a core melt. The primary coolant blows down through the break to the drywell, depressurizing the RCS. As the pressure builds in the drywell, steam and other gases expand into the wetwell. Passing through the suppression pool water, the steam is condensed, thereby reducing the pressure in the wetwell and drywell. A reactor trip occurs and the ECCS actuates to remove fuel decay heat. Thermodynamic analyses, performed using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis conservatively assumes that ECCS is not effective and that substantial fuel damage occurs. For these analyses, the failure of the largest pipe in the RCS is postulated as this condition represents the larger challenge to mitigating the radio nuclide releases. Appendix A of RG 1.183 identifies acceptable radiological analysis assumptions for a LOCA.

3.1.2.1 Source Term

The licensee projected the core inventory of fission products using the ORIGEN-S computer code. The ORIGEN-S computer code is acceptable to the NRC staff for estimating the core inventory. The licensee assumed a core licensed power level of 3430 megawatts thermal (MWt) and applied an uncertainty correction of 102 percent to arrive at the analysis input of 3499 MWt. The resulting core inventories of dose-significant radionuclides were tabulated in Table 1 to Enclosure 1 of the February 13, 2003, application.

Fission products from the damaged fuel are released into RCS and then into the primary containment (i.e., drywell and wetwell). For a LOCA, it is anticipated that the initial release to the primary containment will last 30 seconds and that all of the radioactive materials dissolved or suspended in the RCS liquid will be released to the containment. The gap inventory release phase begins 2 minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. These data are summarized in the table attached to this SE. The inventory in each release phase is released at a constant ramp starting at the onset of the phase and continuing over the duration of the phase. The NRC staff finds that these assumptions are consistent with the guidance of RG 1.183 and are, therefore, acceptable.

3.1.2.2 Release Pathways

Once dispersed in the primary containment, the release to the environment is assumed to occur through three pathways:

- Design leakage of primary containment atmosphere;
- Leakage of primary containment atmosphere via design leakage through MSIVs; and
- Design leakage from ECCS piping and components that recirculate suppression pool water outside the primary containment.

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the two recirculation loops resulting in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell. Because the previous Technical Information Document (TID) 14844, dated March 23, 1962, specified a source term assumption of instantaneous core damage, this initial blowdown would also include all of the released fission products, a fraction of which would be retained by the suppression pool water. Under the AST, a substantial fraction of the fission product release occurs after the initial blowdown is complete. As such, the licensee did not credit any reduction in fission products transferred to the torus air space by suppression pool scrubbing, assuming instead a well-mixed torus air space and drywell. The NRC staff finds the licensee's assumptions in this area acceptable.

The licensee assumed that a portion of the fission products released from the RPV will plate out in the containment due to natural deposition processes. The licensee modeled this deposition using the 10-percentile model described in the NRC staff-accepted NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (i.e., the "Powers Model"). The NRC staff finds the licensee's assumptions in this area acceptable.

3.1.2.3 Use of Standby Liquid Control for Suppression Pool pH Control

The AST assumes that the iodine released to the containment includes 95-percent cesium iodide, 4.85-percent elemental iodine, and 0.15-percent organic forms. The assumption of this iodine specification is predicated on maintaining the suppression pool water at pH 7.0 or higher. At pH values less than about 7.0, elemental iodine may evolve from the water pool and invalidate this iodine specification. The licensee proposed to use the SLC system to establish and maintain the pH level of the suppression pool at pH 7.0 or higher. The SLC system is included in boiling-water reactors as a redundant means of establishing a reactor shutdown. For this design function, the SLC system injects sodium pentaborate into the RPV just below the lower core support plate. The sodium pentaborate solution, a neutron poison, establishes and maintains an adequate shutdown margin to keep the reactor subcritical. However, sodium pentaborate also acts as a pH control agent. It is this latter characteristic of sodium pentaborate that the licensee used in assigning a new design function to the SLC system.

Since failure to achieve and maintain the suppression pool pH could increase the fission products available for release from the primary containment, the NRC staff expects the SLC system to have the characteristics of a safety-related system and meet the requirements of GDC-41. The NRC staff requested additional information of the licensee that would show that the SLC system would have redundancy, reliability, quality, and treatment comparable to a safety-related system. The licensee responded by supplemental letters dated June 4, 2004, and July 30, 2004, which documented the following characteristics of the Fermi 2 SLC system:

- The SLC system components necessary for the injection of sodium pentaborate are classified as quality assurance (QA) Level 1M, signifying that, although not originally intended, procured, designed, or classified as safety-related, the SLC system is being maintained and tested as a safety-related system.

- The SLC system is provided with backup power supplied by the emergency diesel generators.

- The SLC system components essential for injecting the sodium pentaborate have been shown to be able to withstand the effects of a safe shutdown earthquake in accordance with the Fermi 2 licensing basis.

- The SLC system is classified as a risk-significant system by the Fermi 2 maintenance rule program and is included in the in service inspection (ISI) and in service testing (IST) programs.

- The SLC system is not included in the Fermi 2 environmental qualification program. However, the injection of sodium pentaborate would be completed before the environmental conditions at the SLC component locations outside the primary containment exceeded the Fermi 2 harsh area criteria. The only component located within a harsh area is the SLC injection line inboard check valve. However, the valve sub-component subject to degradation (i.e., a soft seat) is not required for the opening function associated with injection.

The Fermi 2 SLC system active components, with three exceptions, are redundant and can be expected to support sodium pentaborate injection assuming a single failure. The three nonredundant active components are (1) the SLC injection line inboard check valve, (2) the SLC injection outboard check valve, and (3) the control room initiation switch. The injection line inboard and outboard check valves are in series and failure of either could block injection. The control room initiation switch has redundant control sections on a common shaft. The operator selects which division is to be actuated. The licensee provided design, procurement, testing, and performance data that shows these components have acceptable quality and reliability to compensate for the lack of redundancy.

The injection check valves are tested in both the open and closed directions every 18 months. The licensee's search of the plant corrective action database and an industry-wide database did not identify any failures that would have prevented sodium pentaborate injection. The discharge pressure of the SLC pumps (positive displacement) creates a significant opening force on the valve disk. Generic performance data used in probabilistic risk assessments estimate the probability of a check valve failing to open as approximately once in 3,700 attempts. The outboard check valve can be bypassed with a temporary hose; the inboard check valve would be inaccessible. If the control room initiation switch fails to start either division, temporary jumpers can be installed in the relay room to start a pump and actuate the explosive valves.

The SLC system operating procedure and the alarm response procedure for the containment high area radiation monitor were developed and maintained in accordance with the quality assurance requirements of 10 CFR Part 50, Appendix B. The containment high area radiation monitors meet the requirements of Category 1 instruments as defined in Tables 1 and 2 of NRC RG 1.97 for Type E variables.

The NRC staff considered the path for ECCS flows through the RPV. Low pressure core injection (LPCI) discharges into the reactor recirculation discharge loop for injection into the core via the jet pumps. At Fermi 2, a LPCI loop select control logic selects the unbroken loop and all of the LPCI flow (20,000 gpm per division) is directed to that loop (the discharge to the broken loop is blocked). The LPCI flow is directed into the lower-head region through 10 jet pump nozzles. These jet pump nozzles accelerate the LPCI flow and drive it into the lower core region. The flow is driven out of the lower head through the nonactive jet pumps in the reverse direction, spilling out of the broken recirculation loop. The LPCI loop select prevents LPCI from bypassing the RPV via the broken recirculation loop. In addition, the separate core spray (CS) system pushes water down through the reactor core at a flow rate of about 6000 gpm per division. The 26,000 gpm of water being forced through the RPV exceeds the reactor boil off rate of 600 gpm. The LPCI and CS flow and the relatively small SLC injection flow provides for good mixing in the RPV and the transport of the sodium pentaborate from the RPV to the suppression pool via spillage of greater than 25,000 gpm. The licensee estimates than in the 6 hours from the start of the accident, the RPV and attached piping would be turned over more than 50 times and the suppression pool water would be turned over more than 7 times. This assures complete mixing and provides adequate justification for the dose analysis assumption of effective suppression pool chemistry control at 6 hours.

The NRC staff finds that the SLC system, as used for post-LOCA pH control, has suitable redundancy in its components and features to assure that a suppression pool pH of 7.0 or greater can be achieved within 24 hours and maintained for 30 days post-LOCA. As such, the NRC staff finds that the proposed new safety function appropriately satisfies the criteria of GDC-41. The NRC staff bases this decision on the redundancy, reliability, quality, maintenance and testing treatment for this system. Although a single failure of one of the injection check valves or the initiation switch could prevent injection of the sodium pentaborate solution, the NRC staff finds that the performance history of these components shows a suitably high degree of reliability and that the maintenance and testing program for these components provide reasonable assurance of continued reliability. The NRC staff also notes there would be adequate time before the 24-hour pH control criterion to affect alternative injection paths. In this regard, the NRC staff notes that the Fermi 2 emergency operating procedures provide instructions for an alternative injection path via the standby feedwater system. The NRC staff also finds that the ECCS flow paths in the RPV provide reasonable assurance that the injected sodium pentaborate solution will mix with the suppression pool contents in sufficient quantities to achieve and maintain the pH level.

Section 4.0 of this SE contains details of regulatory commitments made by the licensee regarding the operation of the SLC system.

3.1.2.4 Containment Leakage Pathway

The drywell and wetwell are projected to leak at their design leakage of 0.5 percent of their atmospheric contents by weight per day for the first 24 hours and 0.25 percent of their atmospheric contents by weight for the remainder of the 30-day accident duration. Leakage from the drywell and wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. This pressure differential ensures that leakage from the drywell and wetwell is collected and processed by the SGTS. The SGTS exhaust is processed through HEPA filter media before release to the environment. The licensee states that there would be a 17-minute period from the start of the event during which the secondary containment may not be at a negative pressure. Therefore, although SGTS is assumed to actuate on high drywell pressure early in the event, SGTS filtration is not credited for the first 17 minutes. Five percent of the stated primary containment leak rate is assumed to bypass the secondary containment and enter the environment as a ground level release. The licensee did not assume any release transport mitigation (e.g., deposition) within the bypass pathways. This bypass is not applicable to the MSIV or ECCS leakage pathways. These assumptions are consistent with the guidance of RG 1.183 and are, therefore, acceptable to the NRC staff.

3.1.2.5 Main Steam Isolation Valve Leakage

The four main steam lines, which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are three MSIVs on each steam line, one inside containment (i.e., inboard), one outside containment (i.e., outboard), and one located before the combined header in the turbine building. The main steam lines are seismically qualified through the third MSIV, from the RPV. The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release.

During a LOCA, fission products released from the core via the severed recirculation line are dispersed equally throughout the drywell. Following initial blowdown of the RPV, steam produced in the RPV carries fission products to the drywell. When core cooling is restored, steam is rapidly generated in the core. This steam and the ECCS flow carry fission products from the core to the drywell via the severed recirculation line, resulting in well-mixed RPV dome and drywell fission product concentrations. Once the rapid steaming stops, the drywell contents can flow back into the RPV through the severed line and are available for release via the

MSIVs. The licensee described its analysis of this release path in its February 13, 2003, application. In response to NRC staff's concerns, the licensee reanalyzed this release pathway, as described in the July 30, 2004, supplemental letter.

The licensee's revised analysis assumes that the released fission products are dispersed throughout the drywell free volume, that there is no mixing of the drywell and wetwell volumes for the first 2 hours, and that there is complete mixing after that. The NRC staff finds this assumption acceptable as the AST is effectively based on a terminated LOCA in which core cooling is restored at the end of the early in-vessel release phase. The licensee assumes an MSIV leak rate of 100 scfh in steam line B, 50 scfh in steam line D, and no leakage in remaining steam lines A and C for the first 24 hours and 50 percent of these values for the next 29 days. Steam lines B and D are each modeled as two deposition nodes – the RPV nozzle to the inboard MSIV, and the inboard MSIV to the third MSIV. Since a main steam rupture within the primary containment could be an initiator for a LOCA, the licensee assumes no deposition credit for the inboard node in steam line B. In keeping with single failure considerations, the outboard MSIV on steam line B is assumed to fail open.

The inboard steam line piping is assumed to remain at a temperature of 554 °F for the duration of the accident. The outboard steam line piping is assumed to be 554 °F for the first 8 hours and decrease in four steps to 289 °F at 96 hours. The inboard piping segments are assumed to be at 25 psig, the MSIV test pressure; the outboard segments are assumed to be at atmospheric pressure. Post-accident drywell pressure will exceed the MSIV test pressure for the first 10 minutes. However, the assumption of a constant leakage rate for 24 hours, the assumption that the inboard segments are at the 100-percent power steam line temperature for the entire accident duration, and the relatively low magnitude of the fission product release during the gap-release phase of the accident adequately compensate for the underestimation of the flow rate in this short period. Since the TS-allowable leakage is assessed in units of scfh and the steam lines are not at standard conditions of temperature and pressure, the licensee adjusts the assigned flow rates appropriately.

The licensee modeling assumes well-mixed deposition nodes. For aerosol deposition, the volumes and 50 percent of the internal surface area of the horizontal piping runs are input to the deposition model. For elemental radioiodine deposition, all of the volume and internal surface area of piping runs are input to the deposition model. The licensee did not credit deposition of organic iodine. The deposition of aerosols was modeled as described in Appendix A of AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term." The deposition of elemental iodine was modeled using the same formulations, but using the deposition velocities obtained using a correlation used in the RADTRAD code. (See Equation 30 in NUREG/CR-6604.)

The licensee modeled the steam line deposition and assigned model parameter values and assumptions in an appropriate deterministic manner that conservatively reduces the amount of deposition credit obtained. The NRC staff finds that this model is consistent with the broad guidance in Appendix A of RG 1.183 and is, therefore, acceptable.

3.1.2.6  Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS and by natural processes such as deposition and plate-out.  Post-LOCA, the suppression pool is a source of water for the ECCS.  Since portions of these systems are located outside the primary containment, potential leakage from these systems is evaluated as a radiation exposure pathway.  For the purposes of assessing the consequences of leakage from the ECCS, the licensee conservatively assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool.  Noble gases are assumed to remain in the drywell atmosphere.  Since aerosols and particulate radionuclides (other than iodine) will not become airborne on release from the ECCS, they are not included in the ECCS source term.  This source term assumption is conservative in that all of the radioiodine released from the fuel is credited in both the primary containment atmosphere leakage and the ECCS leakage.  In a suitable mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time.  The analysis considers the equivalent of 5 gpm unfiltered ECCS leakage starting at the onset of the LOCA.  The licensee assumes that 2 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97-percent elemental and 3-percent organic iodine.  As assumed for the primary containment leakage pathway, the leakage enters the environment via the SGTS as a filtered release.  The release continues for 30 days.  The NRC staff finds these assumptions to be consistent with the guidance of RG 1.183 and, therefore, acceptable.

3.1.2.7  Offsite Doses

The licensee evaluated the maximum 2-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ.  The resulting doses are less than the 10 CFR 50.67 criteria and are, therefore, acceptable.

3.1.2.8  Control Room Doses

The licensee evaluated the dose to the operators in the control room.  The control room emergency filtration (CREF) system is automatically actuated by a primary containment isolation signal, by high radiation at the control room outside air intakes, or by manual actuation by the control room operators.  Since the primary containment isolation actuation is triggered by plant process sensors, such as RPV low water level and high drywell pressure, isolation of the control room is assumed to be immediate (i.e., completed before substantial fission products are released) for the LOCA.  During normal operations, the unfiltered outside air makeup to the control room ventilation system intake is 4000 cfm.  Once isolation occurs, an 1800 cfm emergency makeup air flow is mixed with 1200 cfm of recirculation flow rate, passed through the control room emergency filters, and introduced to the control room.

Although the control room is designed to be pressurized during an accident event, for the LOCA analysis,  the licensee assumes that there is 600 cfm of unfiltered inleakage into the control room envelope (CRE).  For the FHAs, no credit is taken for CRE isolation.  The 600 cfm unfiltered inleakage assumption is not directly based on results from the performance of integrated testing, but rather, is an arbitrarily large value that the licensee expects will bound the actual leakage measured by testing.

On June 12, 2003, the NRC staff issued Generic Letter 2003-01, "Control Room Habitability." This generic letter identifies NRC staff concerns regarding the reliability of current surveillance testing to identify and quantify control room inleakage, and requests licensees to confirm the most limiting unfiltered inleakage into their CRE. The February 13, 2003, application predates the generic letter. On August 11, 2003, the licensee submitted a "60-day" response to this generic letter and followed up on December 8, 2003, with a "180-day" response. The NRC staff is currently evaluating these responses separately from the February 13, 2003, application. In the December 8, 2003, letter, the licensee committed to provide results of the tracer gas testing to the NRC by December 9, 2004. However, this commitment was predicated on the NRC staff's approval of the February 13, 2003, application before the preliminary test date of October 6, 2004. The licensee revised this commitment in a new license amendment request dated July 30, 2004, in which the licensee proposes to add a condition to the Fermi 2 license that requires the tracer gas test be completed before March 31, 2005. The July 30, 2004, application is currently under NRC staff review.

As noted above, the licensee assumed an unfiltered inleakage value of 600 cfm in its analyses. The NRC staff has determined, with reasonable assurance, that the Fermi 2 control room will be habitable during a LOCA and FHAs and that this amendment may be approved before the completion of the tracer gas testing and the final resolution of the generic issue without an adverse impact on adequate public protection. The NRC staff bases this determination on (1) the relative amount of the infiltration assumed in the licensee's analyses, (2) the relatively low unfiltered inleakage indicated by periodic testing of duct work leakage, and (3) the NRC staff's experience to date that duct work leakage is generally (but not always) a significant portion of the overall CRE unfiltered inleakage.

In Table 5 of Enclosure 1 of its February 13, 2003, application, the licensee provided tabulated CRE flow rates. The CRE volume of 252,000 $ft^3$ is larger than the actual control room volume of 57,000 $ft^3$. The licensee proportioned the CREF air intake, recirculation, and unfiltered inleakage to the CRE to obtain a value for the control room "shine" volume. Thus, rather than 600 cfm of unfiltered inleakage into the CRE, the licensee tabulated a proportioned value of 135.3 cfm for unfiltered inleakage with other flows similarly proportioned. Although no scrutable basis exists for such an apportioning, the modeling approach assumes an instantaneous, homogeneous distribution. As such, the reduced inlet and outlet flow rates and reduced volume yields the same air concentrations (µCi/cc) as if the higher flow rates and higher volumes were used. Although the inhalation doses will be unchanged, the external shine dose may be less since the finite cloud correction applied to photon exposures will be based on a smaller cloud size. The NRC staff's independent calculations showed that the inhalation doses comprise a substantial fraction of the overall doses.

3.1.2.9 LOCA Summary

Based on the NRC staff's review of the licensee's analysis and independent confirmatory calculations performed by the NRC staff, the NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the LOCA are acceptable. The table attached to this SE provides selected assumptions found acceptable to the NRC staff.

### 3.1.3 Fuel Handling Accidents

This accident analysis postulates that a spent fuel assembly is dropped during refueling. The kinetic energy developed in this drop is conservatively assumed to be dissipated in the mechanical damage to the cladding on several fuel rods. The fission product inventory in the core is largely contained in the fuel pellets that are enclosed in the fuel rod clad. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the pool water, depending on their physical and chemical form. The fission products released from the pool are assumed to be released to the environment without credit for reactor building holdup or dilution via the refueling building ventilation system. The control room was modeled without taking credit for automatic system actuation. Therefore, the normal outside air makeup flow of 4000 cfm continues for the duration of the event and no credit is taken for CREF filters.

Fermi 2 License Amendment No. 144 approved selective implementation of an AST to the FHA for certain fuel types and fuel burnups. Three cases were considered:

- GE14 10 x 10 fuel that meets RG 1.183 guidance in Footnote 11[1]

- GE11 9 x 9 fuel that meets RG 1.183 guidance in Footnote 11

- GE11 9 x 9 fuel that does not meet RG 1.183 guidance in Footnote 11

The AST implementation authorized by License Amendment No. 144 was limited to the first two cases. For the third case, the previous TID 14844 source term and the whole-body and thyroid dose limits were retained. Each of these three cases was further subdivided into two subcases: SGTS available and SGTS not available.

In the February 13, 2003, application the licensee proposed continuing the approach approved in License Amendment No. 144. The NRC staff agrees to the continuance of the licensing bases established by License Amendment No. 144 in conjunction with the program commitments made by the licensee to ensure that the RG 1.183 guidance in Footnote 11 is satisfied. Selected assumptions found acceptable to the NRC staff are presented in the table attached to the SE. The NRC staff did independent calculations and confirmed the licensee's conclusions. The NRC staff finds the EAB, LPZ, and control room doses estimated by the licensee for the FHA cases to be acceptable.

### 3.1.4 Atmospheric Relative Concentrations

Atmospheric relative concentration ($\chi$/Q) values are a measure of the amount of diffusion and transport of effluents released to the environment during DBAs. In support of this amendment request, the licensee calculated $\chi$/Q values for the EAB, LPZ, and control room. The offsite $\chi$/Q values were determined using the guidance of RG 1.145, "Atmospheric Dispersion Models for

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[1] Footnote 11 of RG 1.183 provides specific limitations for the application of RG 1.183 release fractions. These limitations include burnup and linear heat generation restrictions.

Potential Accident Consequence Assessments at Nuclear Power Plants," and the NRC-sponsored PAVAN computer code. In a change from the original design basis, all wind directions, including those offshore, were considered. Control room $\chi$/Q values were calculated using the NRC-sponsored ARCON96 computer code.

In support of License Amendment No. 144, the licensee submitted hourly meteorological data observations for the Fermi 2 site for the years 1995 to 1999. The NRC staff used this data in doing independent calculations to confirm the licensee's values. Based upon its review of the data and the results of confirmatory calculations, the NRC staff finds the $\chi$/Q values tabulated in Tables 11 and 12 of Enclosure 1 to the February 13, 2003, application, to be acceptable.

3.1.5 Other Radiological Consequence Analyses

The licensee has considered the impact of the proposed amendment on its commitments with regard to equipment qualification. Consistent with guidance in RG 1.183, the previous analyses performed using a source term that is consistent with the previous commitments under 10 CFR 50.49 remain effective. Similarly, the licensee evaluated the impact of the proposed changes on the post-accident dose rates to vital plant areas used in post-accident operations. The NRC staff finds that these areas remain accessible given the proposed changes and are, therefore, acceptable.

3.2 <u>Proposed Design-Basis and TS Changes</u>

The following sections discuss design-basis and TS changes proposed by the licensee.

3.2.1 Design-Basis Change

The licensee proposed to modify the Fermi 2 design basis to selectively replace the current accident source term with an AST and replace the previous whole-body and thyroid accident dose guidelines with the TEDE criteria of 10 CFR 50.67(b)(2) for LOCA and FHAs.
The licensee originally proposed a full implementation of the AST consistent with the guidance provided in RG 1.183, but subsequently reduced the amendment scope to a selective implementation involving only the LOCA and FHAs. Selective use of an AST at Fermi 2 for FHAs had been previously approved in two separate amendments (License Amendment Nos. 143 and 144). Proposed changes in the Fermi 2 design basis are: (1) an increase in assumed control room envelope inleakage; (2) increased main steam isolation valve (MSIV) leakage; (3) development of new offsite and control room atmospheric relative concentrations ($\chi$/Q); and (4) the assignment of a new design function to the SLC system. Based on the above evaluation, the NRC staff finds the proposed changes acceptable.

3.2.2 TS 3.6.1.3, "Primary Containment Isolation Valves"

The licensee proposed to (1) increase the allowable secondary containment bypass leakage specified in SR 3.6.1.3.11 from less than or equal to 0.04 times the maximum allowable leakage rate for the primary containment ($L_a$) to an "equivalent" leakage rate of less than or equal to 0.05 times the maximum allowable leakage rate for the primary containment ($L_a$), and (2) increase the MSIV allowable leakage specified in SR 3.6.1.3.12 from less than or equal to 100 scfh for all four steam lines to less than or equal to 150 scfh in all four steam lines and less than or equal to 100 scfh for any one steam line.

As discussed above, the licensee proposed to incorporate acceptable models for crediting fission product removal in the main steam lines. This modeling was found acceptable to the NRC staff. The licensee also proposed to use this modeling in crediting deposition in the other bypass pathways, but did not include these corrections in the dose analysis. Instead, the licensee proposed to apply such corrections to the surveillance test results by modifying SR 3.6.1.3.11 to refer to “equivalent” leakage. The licensee did not provide sufficient information for the NRC staff to make a determination regarding its acceptability. By letter dated September 16, 2004, the licensee withdrew this portion of the proposed change to SR 3.6.1.3.11.

The proposed replacement SR values for TS 3.6.1.3 were used in the reanalysis of the LOCA radiological consequence analyses done in support of this amendment request. These proposed changes are addressed in the analyses found acceptable above, and are therefore acceptable.

3.2.3 TS 3.6.1.9, “Main Steam Isolation Valve Leakage Control System”

The licensee proposed to delete TS 3.6.1.9 since operability of the system is not assumed in any accident analysis.

The design function of the MSIV leakage control system (LCS) is to supplement the isolation function of the MSIVs by containing the fission products that could leak through the closed MSIVs after a LOCA. The MSIV LCS is designed to operate at a higher pressure than the containment after a postulated LOCA to provide protection against potential leakage of radioactive contaminants to the environment. In its submittal, the licensee stated that the elimination of the requirements for MSIV LCS operation would result in reduced personnel exposure during maintenance on the system. In addition, eliminating the MSIV LCS function may reduce the demand requirements on the non-interruptible air supply, thereby increasing margins for operation of this safety system.

The bases for the NRC staff’s acceptance of the proposed TS deletion are that the radiological consequences calculated by both the licensee and the NRC staff for the postulated LOCA without crediting MSIV LCS operation are within the dose acceptance criteria specified in 10 CFR 50.67, and the methods and major parameters and assumptions used in the licensee’s dose calculations are consistent with the guidelines provided in RG 1.183 and the NRC staff’s technical positions. The function of the MSIV LCS is no longer credited in the mitigation or prevention of any design basis accident and therefore, the MSIV LCS no longer meets the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. Therefore, the NRC staff finds the proposed change to delete TS 3.6.1.9 to be acceptable.

3.2.4 TS 3.6.4.1, “Secondary Containment”

The licensee proposed to increase the allowed secondary containment draw-down time from 567 seconds to 12 minutes in SR 3.6.4.1.5. The proposed replacement SR value is less than the draw-down time assumed in the above analysis of the LOCA (i.e., 17 minutes). Based on the above evaluation, the NRC staff finds the change to TS 3.6.4.1 acceptable.

3.2.5 TS Bases

The proposed changes to the above TSs require that the licensee revise the discussion in the associated TS Bases section. Although the licensee's application included possible wording for the revised TS Bases, the licensee will formally address the change to the Bases in accordance with the Fermi 2 Bases Control Program. The TS Bases should be addressed separately from this amendment and should be included in a future update of the TS Bases in accordance with the Fermi 2 Bases Control Program.

## 4.0 REGULATORY COMMITMENTS

In its supplemental letter of June 4, 2004, the licensee made the following commitments that are to be implemented concurrently with the implementation of this amendment:

1. The SLC System Operating Procedure will be revised to describe the new system function of controlling suppression pool pH post-LOCA.

2. The Alarm Response Procedure for the Containment High Area Radiation Monitor will be revised to direct operators to initiate SLC when high radiation levels and LOCA symptoms are detected in the primary containment.

3. The plant Emergency Operating Procedures will be revised to clearly direct operators to maintain SLC injection when it is required for suppression pool water pH control.

4. Operator training will be updated to reflect the pH control function of the SLC system.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program (See Regulatory Issue Summary 2000-017, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff"). The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

## 5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 28847). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).  Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 7.0 CONCLUSION

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

Attachment:  Table

Principal Contributor:  S. F. LaVie

Date: September 28, 2004

## Table

## Fermi 2 Selected Accident Analysis Parameters

### General

| | |
|---|---|
| Reactor power (3430 x 1.02), MWt | 3499 |
| Core Inventory | Table 1, February 13, 2003, application |
| Dose conversion factors | FGR11/FGR12 |
| Breathing rate, offsite, $m^3/s$ | |
| 0-8 hours | 3.5E-4 |
| 8-24 hours | 1.8E-4 |
| >24 hours | 2.3E-4 |
| Breathing rate, control room, $m^3/s$ | 3.5E-4 |
| Control room normal intake flow, cfm | 4000 |
| Control room unfiltered infiltration, cfm | 600 |
| Control room filtered pressurization, cfm | 1800 |
| Control room filtered recirculation, cfm | 1200 |
| Control room volume, $ft^3$ | 252,000 |
| Control room charcoal filter efficiency*, % | 95 |
| *intake also passes through recirculation filter | |
| Control room recirculation charcoal filter efficiency, % | 95 |
| Control room occupancy factor | |
| 0-24 hrs | 1.0 |
| 1-4 days | 0.6 |
| 4-30 days | 0.4 |
| SGTS Filter Efficiency, % | 99 |

**Offsite $\chi/Q$ values, $sec/m^3$**

| Time Period | EAB | LPZ |
|---|---|---|
| 0 - 2 hours | 2.09E-4 | 4.86E-5 (FHA) |
| 0 - 8 hours | – | 2.17E-5 (LOCA) |
| 8 - 24 hours | – | 1.45E-5 (LOCA) |
| 24 - 96 hours | – | 6.02E-6 (LOCA) |
| 96 - 720 hours | – | 1.71E-6 (LOCA) |

**Control Room $\chi/Q$ values, $sec/m^3$**

| Time Period | SGTS | MSIV | FHA w/o SGTS | FHA w SGTS |
|---|---|---|---|---|
| 0-7.2 sec | | | | 4.03E-3 |
| 0 - 2 hours | 6.18E-4 | 3.10E-4 | 4.25E-3 | 3.65E-3 |
| 2 - 8 hours | 4.53E-4 | 2.33E-4 | | |
| 8 - 24 hours | 1.88E-4 | 9.93E-5 | | |
| 24 - 96 hours | 1.26E-4 | 7.08E-5 | | |
| 96 - 720 hours | 8.70E-5 | 5.48E-5 | | |

## Loss-of-Coolant Accident

### Containment Leakage Source

| | |
|---|---|
| Onset of gap release phase, min | 2.0 |

**Core release fractions and timing–Containment atmosphere**

| Duration, hrs | 0.5000E+00 | 0.1500E+01 |
|---|---|---|
| Noble Gases: | 0.5000E-01 | 0.9500E+00 |
| Iodine: | 0.5000E-01 | 0.2500E+00 |
| Cesium: | 0.5000E-01 | 0.2000E+00 |
| Tellurium: | 0.0000E+00 | 0.5000E-01 |
| Strontium: | 0.0000E+00 | 0.2000E-01 |
| Barium: | 0.0000E+00 | 0.2000E-01 |
| Noble Metals: | 0.0000E+00 | 0.2500E-02 |
| Cerium: | 0.0000E+00 | 0.5000E-03 |
| Lanthanum: | 0.0000E+00 | 0.2000E-03 |

**Iodine species distribution**

| | |
|---|---|
| Elemental | 0.95 |
| Organic | 0.0485 |
| Particulate | 0.0015 |

**Primary containment volume, $ft^3$**

| | |
|---|---|
| Drywell | 163,730 |
| Suppression pool air space | 130,900 |

**Containment leak rate, %/day**

| | |
|---|---|
| 0- 24 hours | 0.5 |
| Greater than 24 hours | 0.25 |

| | |
|---|---|
| Secondary containment bypass, % | 5.0 |
| Secondary containment volume*<br>*no mixing/holdup credited | 2.80E6 |
| SGTS draw down time, min | 17 |

**Drywell natural deposition**

| | |
|---|---|
| Particulate | Powers 10-percentile Model |
| Elemental | None |

| | |
|---|---|
| Control room isolation delay, minutes | 0 |

## Main Steam Isolation Valve Leakage

Activity same as containment leakage case dispersed in drywell volume only

**MSIV TS leak rate at 25 psig, scfh***

| | |
|---|---|
| One line | 100 |
| Total | 150 |

*reduced 50 percent after 24 hours

**Steam line (and steam) temperature, °F**

| | Inboard | Outboard |
|---|---|---|
| 0-8 hrs | 554.0 | 554.0 |
| 8-16 | 554.0 | 510.7 |
| 16-24 | 554.0 | 472.1 |
| 24-96 | 554.0 | 437.7 |
| 96-720 | 289.4 | 289.4 |

**Main steam line configuration for deposition analysis**

- No credit for deposition beyond third MSIV
- No credit for holdup or plate-out in the main condenser
- All four steam lines intact, in service at start of event
- 100 scfh assigned to steam line B; 50 scfh to line D
- Two well-mixed deposition nodes in each steam line: RPV to inboard MSIV, inboard MSIV to third MSIV
- Inboard steam line B is ruptured; no deposition credit
- Steam line B outboard MSIV fails to close
- Only horizontal lines are credited for aerosols
- Pressure in inboard segments is assumed to be equal to test pressure
- Pressure in outboard segments is assumed to be atmospheric
- AEB-98-03 Appendix A well-mixed aerosol deposition model
- AEB-98-03 formulation used with deposition velocity from RADTRAD Bixler correlation

Release via Turbine Building HVAC Exhaust

## ECCS Leakage

**Iodine species fraction**

| | |
|---|---|
| Particulate/aerosol | 0 |
| Elemental | 97 |
| Organic | 3 |

| | |
|---|---|
| Suppression pool liquid volume, gal | 949,200 |
| Estimated leakage, gpm | 5 |
| Iodine Flash Fraction | 0.02 |
| SGTS charcoal filtration | Credited |

Release via SGTS

## Fuel Handling Accident

| | |
|---|---|
| Core peaking factor | 1.7 |
| **Fuel rods damaged, rods** | |
| GE14 fuel | 172 |
| GE11 fuel | 140 |
| **Decay period, hrs** | |
| GE14 fuel (no SGT) | 151 |
| GE11 fuel (no SGT) | 107 |
| Either with SGT | 24 |
| **Fraction of core in gap** | |
| I-131 | 0.08 |
| Kr-85 | 0.1 |
| Other iodines | 0.05 |
| Other noble gases | 0.05 |
| Alkali Metals | 0.12 |
| Pool decontamination factor | 200 |
| Release period, hr | 2 |
| Refuel floor volume, $ft^3$ | 950,000 |
| Ventilation flow rate, cfm | 95,000 |
| **Release location** | |
| SGT operating | SGT stack |
| No SGT | RB vent stack |
| Control room CREF initiation | Not Credited |
| Delay in SGT filter actuation, sec | 7.2 |