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DTE Energy

10 CFR 50.90
10 CFR 50.67

June 29, 2005
NRC-05-0045

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 Request to Revise the Combined Secondary Containment Bypass Leakage Rate and the Combined MSIV Leakage Rate Limits in Technical Specification 3.6.1.3

Pursuant to 10 CFR 50.90 and 10 CFR 50.67, Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License, Appendix A, Technical Specifications (TS) to revise Surveillance Requirements (SR) 3.6.1.3.11 and 3.6.1.3.12 in TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)." Specifically, this proposed amendment would revise the combined secondary containment bypass leakage rate limit for all bypass leakage paths in SR 3.6.1.3.11 from 0.05 to 0.10 L_a and the combined Main Steam Isolation Valve (MSIV) leakage rate limit for all four main steam lines in SR 3.6.1.3.12 from 150 to 250 Standard Cubic Feet per Hour.

The proposed amendment is based on a revision to the analysis of the design basis Loss of Coolant Accident (LOCA) in accordance with the Alternative Source Term (AST) methodology. In addition to the changes in the TS leakage rate limits, the revision to the analysis includes a reduction to the design basis Control Room Envelope (CRE) unfiltered inleakage from 600 to 300 Cubic Feet per Minute. This reduction in the CRE unfiltered inleakage design basis is based on the results of the baseline tracer gas testing of the Fermi 2 CRE performed in March 2005.

The revision to the LOCA design basis analysis performed in support of this license amendment request utilizes the same methodology previously approved by the NRC in license amendment number 160. The NRC has also approved a similar change to the design basis CRE unfiltered inleakage based on the results of the baseline tracer

A001

gas testing for the Hope Creek Generating Station (ADAMS Accession Number ML030760293).

Enclosure 1 provides an evaluation of the proposed license amendment, including an analysis of the issue of significant hazards consideration using the standards of 10 CFR 50.92. Detroit Edison has concluded that the change proposed in this submittal does not result in a significant hazards consideration. Enclosure 2 provides a marked up page of the existing TS to show the proposed change. Enclosure 3 provides a clean version of the affected TS page with the proposed change incorporated. Enclosure 4 provides a copy of marked up TS Bases pages affected by this change. Enclosure 5 provides the RADTRAD computer program input files and nuclide information files. Enclosures 4 and 5 are provided for information only.

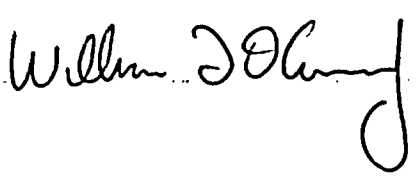
Detroit Edison has reviewed the proposed change against the criteria of 10 CFR 51.22 and has concluded that it meets the criteria provided in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

Detroit Edison requests NRC approval of this proposed license amendment by January 3, 2006, with an implementation period of within 60 days following NRC approval. The requested approval date is based on the need for applying the revised leakage limit rates during the upcoming eleventh refueling outage scheduled to start in March 2006.

No new commitments are being made in this letter.

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

Sincerely,



cc: D. P. Beaulieu
E. R. Duncan
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 29th day of June, 2005 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.

CYNTHIA L. SMITH
Notary Public, Monroe County, MI
My Commission Expires Oct 5, 2005

Notary Public

ENCLOSURE 1
TO NRC-05-0045

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

REQUEST TO REVISE THE SECONDARY CONTAINMENT BYPASS LEAKAGE AND MSIV LEAKAGE RATE LIMITS IN SR 3.6.1.3.11 AND SR 3.6.1.3.12

EVALUATION OF THE PROPOSED LICENSE AMENDMENT REQUEST

Evaluation of the Proposed License Amendment Request

Subject: Revision of the Secondary Containment Bypass Leakage and MSIV Leakage Rate Limits in SR 3.6.1.3.11 and SR 3.6.1.3.12

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1.0 Description

Detroit Edison is requesting NRC approval of this proposed revision to the Fermi 2 Technical Specification (TS) Number 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)." The proposed amendment would revise the combined secondary containment bypass leakage rate limit for all bypass leakage paths in Surveillance Requirement (SR) 3.6.1.3.11 from 0.05 to 0.10 of the maximum allowable containment leakage rate (L_a). The proposed amendment would also revise the combined Main Steam Isolation Valve (MSIV) leakage rate limit for all four steam lines in SR 3.6.1.3.12 from 150 to 250 Standard Cubic Feet per Hour (scfh). No change is proposed to the 100 scfh leakage rate limit for any one steam line in SR 3.6.1.3.12.

These changes to the leakage rate limits are based on a revised radiological analysis of the design basis Loss of Coolant Accident (LOCA) in accordance with the Alternative Source Term (AST) methodology. In addition to the change in leakage rate limits for the secondary containment bypass and MSIVs, the revised analysis includes a change to the design basis value for the Control Room Envelope (CRE) unfiltered inleakage from 600 to 300 cubic feet per minute (cfm).

2.0 Proposed Change

The current requirements limiting secondary containment bypass leakage and MSIV leakage are included in TS 3.6.1.3, "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 less than or equal to 0.05 L_a when pressurized to greater than or equal to 56.5 psig.

SR 3.6.1.3.12 requires the verification that the combined MSIV leakage rate for all four main steam lines is less than or equal to 150 scfh and less than or equal to 100 scfh for any one steam line when tested at greater than or equal to 25 psig.

This proposed license amendment would revise the combined secondary containment bypass leakage rate limit for all bypass leakage paths in SR 3.6.1.3.11 from 0.05 to 0.10 L_a. This proposed license amendment would also revise the MSIV leakage rate limit for all four steam lines in SR 3.6.1.3.12 from 150 to 250 scfh. The 100 scfh leakage rate limit for any one steam line in SR 3.6.1.3.12 remains unchanged.

In addition to the two TS leakage rate limit changes mentioned herein, this proposed amendment would revise the design basis limit for the CRE unfiltered inleakage from 600 to 300 cfm. The CRE unfiltered inleakage limit is one of the input parameters assumed in the radiological analysis of the design basis LOCA and is described in the Updated Final Safety Analysis Report (UFSAR).

3.0 Background

The current Fermi 2 licensing basis reflects a selective implementation of the AST methodology in accordance with 10 CFR 50.67 and Regulatory Guide (RG) 1.183 (Reference 1). The design basis LOCA and the Fuel Handling Accident (FHA) are analyzed using AST provisions. All other radiological accidents are analyzed using the TID-14844 (Reference 2) source term.

The NRC has previously approved three license amendments to the Fermi 2 operating license based on the AST methodology. License amendment number 143 (Reference 3) for the revision of the fission product release timing for design basis accidents described in the UFSAR was approved on July 12, 2001. License amendment number 144 (Reference 4) for the reevaluation of the FHA based on AST was approved on September 28, 2001; and license amendment number 160 (Reference 5) for the reevaluation of the LOCA was approved on September 28, 2004.

This proposed license amendment is based on a revision to the radiological analysis of the design basis LOCA. In the original AST LOCA analysis, it was assumed that CRE unfiltered inleakage is equal to 600 cfm for the duration of the accident. This assumption was intended to be a conservative upper bound estimate of the maximum unfiltered inleakage since baseline tracer gas testing of the CRE had not been performed at the time. In Reference 5, the NRC determined that, with the 600 cfm assumption, the Fermi 2 control room will be habitable during a LOCA and that license amendment number 160 may be approved before the completion of the tracer gas testing with no adverse impact on adequate public protection.

As reported in Reference 6, baseline tracer gas testing of the Fermi 2 CRE was performed between March 7 and 10, 2005. The results determined CRE unfiltered inleakage to be 18 cfm (+/- 17 cfm) for Division 1 of the Control Room Emergency Filtration (CREF) System and 59 cfm (+/- 20 cfm) for Division 2. RG 1.197 (Reference 7) does not require the inclusion of uncertainties when reporting inleakage test results for pressurized CREs that demonstrate a CRE inleakage less than 100 cfm. However, even when uncertainties are included, it is clear that the CRE unfiltered inleakage using either one of the two divisions of the CREF System is below 100 cfm. Based on these results, it is concluded that a significant margin is available between the actual values demonstrated by testing and the 600 cfm value assumed in the LOCA radiological analysis approved in Reference 5.

The revised analysis of the LOCA radiological consequences performed to support this proposed license amendment is based on guidance provided in RG 1.183 (Reference 1) and uses identical modeling, methodology, assumptions and computer codes as the analysis approved in license amendment number 160 (Reference 5) with the exception of three input parameters. The CRE unfiltered inleakage is changed from 600 to 300 cfm; the secondary containment total bypass leakage is changed from 0.05 to 0.10 L_a; and the MSIV total leakage for all four steam lines is changed from 150 to 250 scfh.

4.0 Technical Analysis

The analysis of the radiological consequences of a LOCA evaluates 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. Appendix A of RG 1.183 (Reference 1) identifies acceptable radiological analysis assumptions for a LOCA.

The revised analysis of the LOCA radiological consequences performed to support this proposed license amendment is based on RG 1.183. As mentioned above, with the exception of changes in three input parameters, the analysis uses identical modeling, methodology, assumptions and computer codes as the one approved in license amendment number 160 (Reference 5).

4.1 Source Term

The projected core inventory of fission products is derived using the ORIGEN-S computer code (Reference 8). The NRC has accepted the use of the ORIGEN-S computer code for estimating the core inventory. The analysis input of 3499 megawatts thermal (MWt) is based on the current core licensed power level of 3430 MWt with an uncertainty correction factor of 102 percent. The resulting core inventories of dose-significant radionuclides are listed in Table 1. There is no change in the core inventory used in the revised analysis from the one used in the analysis performed to support license amendment number 160.

Fission products from the damaged fuel are assumed to be released into the Reactor Coolant System (RCS) and then into the primary containment. All of the radioactive materials dissolved or suspended in the RCS liquid are assumed to be released to the containment. In accordance with Reference 1, the initial release of the gap inventory phase begins 2 minutes after the event starts and continues for 30 minutes. At that time, the gap inventory release phase ends and the in-vessel release phase begins and continues for 1.5 hours. The core inventory fractions released into containment during each phase are also in accordance with Reference 1. The inventory in each release phase is released at a constant rate starting at the onset of the phase and continuing over the duration of the phase. Table 2 provides key assumptions related to fission product release used in the analysis. The NRC has previously approved these assumptions in license amendment 160 (Reference 5).

4.2 Fission Product Release

The release from the primary containment is assumed to occur through three pathways: design leakage of primary containment atmosphere including secondary containment bypass leakage; leakage of primary containment atmosphere through the MSIVs; and design leakage from ECCS piping and components that recirculate suppression pool water outside the primary containment.

The LOCA considered in the analysis is a complete and instantaneous severance of one of the reactor coolant pressure boundary piping system lines inside containment resulting in a

blowdown of the reactor pressure vessel (RPV) liquid and steam into the drywell. The analysis assumes no mixing between the drywell and torus air space in the initial two hours of the accident. After the two-hour period, the activity in the primary containment is assumed to be instantaneously and homogeneously mixed throughout the drywell and torus air space. No reduction in fission product transfer from the drywell to the torus air space is assumed and no suppression pool scrubbing is credited.

However, credit is taken for natural deposition processes by assuming that a portion of the fission products released from the RPV will plate out in the containment. This deposition is modeled using the 10-percentile "Powers Model" described in NUREG/CR-6189 (Reference 9). The assumptions regarding release pathways in the revised analysis are identical to the ones approved in Reference 5.

4.3 Use of the SLC System 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 7.0, elemental iodine may evolve from the suppression pool water and invalidate this iodine specification. In the original AST LOCA evaluation, Detroit Edison proposed to use the Standby Liquid Control (SLC) System to establish and maintain the pH level of the suppression pool at pH 7.0 or higher. The SLC system injects sodium pentaborate into the RPV just below the lower core support plate. The sodium pentaborate solution acts as a pH buffer and maintains the pool water pH at 7.0 or higher.

In Reference 5, the NRC reviewed the Fermi 2 use of the SLC system for post-LOCA pH control and determined that the system has suitable redundancy in its components and features to assure that a suppression pool pH of 7.0 or greater can be achieved and maintained for 30 days post-LOCA. The NRC found that the pH control safety function appropriately satisfies the criteria of GDC-41 and 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.

The revised LOCA analysis in support of this license amendment does not alter any of the assumptions and procedural provisions established in the previous evaluation approved in Reference 5.

4.4 Containment Leakage Pathway

Table 3 provides key parameters of the primary and secondary containments used in the analysis. The primary containment maximum allowable leak rate is 0.5 percent of its atmospheric contents by weight per day for the first 24 hours. This leak rate is assumed to be reduced by 50 percent for the remainder of the 30-day accident duration. Leakage from the primary containment 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 primary containment is collected and processed by the SGTS. The SGTS exhaust is processed through HEPA filter media before release to the environment. In the analysis, it is assumed 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. These assumptions are the same ones approved in Reference 5.

Ten 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 assumed point of release is the Turbine Building Heating, Ventilation and Air Conditioning (TBHVAC) stack. No deposition or holdup credit is taken within the bypass pathways or the Turbine Building. These assumptions are the same as the ones used in the evaluation approved in Reference 5. The only change is the increase in the allowable secondary containment bypass leakage amount from five to ten percent of L_a.

4.5 Main Steam Isolation Valve Leakage

Table 4 provides key parameters of the main steam line piping leakage and deposition credit assumptions. 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, the inboard MSIV inside containment, the outboard MSIV outside containment, and the third MSIV located before the combined header in the turbine building. The main steam lines are seismically qualified from the RPV through the third MSIVs. The inboard and outboard MSIVs are functionally part of the primary containment boundary. No credit is taken for the third MSIVs. Design leakage through the inboard and outboard MSIVs provides a path for fission products to bypass the secondary containment and enter the environment as a ground level release. The assumed point of release is the TBHVAC stack.

The revised LOCA analysis assumes that the released fission products are dispersed throughout the drywell free volume, that there is no mixing between the drywell and torus air space volumes for the first 2 hours, and that there is complete mixing after that. A conservative distribution of the 250 scfh total MSIV leakage is assumed in the analysis. Table 5 provides a summary of steam line piping geometry and Table 6 provides a summary of steam line nodalization used for iodine deposition evaluation. To minimize deposition credit, the two shortest steam lines (lines B and D) are assumed to each have a flow rate of 100 scfh and the next shortest line (line A) is assumed to have a flow of 50 scfh. These leak rates are assumed constant for the first 24 hours and are reduced to 50 percent of these values for the next 29 days. No flow is assumed in steam line C. Steam lines 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 line rupture within the primary containment could be an initiator for a LOCA, no deposition credit is assumed for the inboard node of steam line B (the longer inboard

pipe segment of the two shortest lines). Also, to satisfy the single failure considerations, the outboard MSIV is assumed to fail open. This is the same assumption used in the evaluation approved in Reference 5.

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. Table 7 provides a summary of the outboard main steam line temperature profile used in the analysis. The inboard piping segments are assumed to be at 25 psig, the MSIV test pressure; and the outboard segments are assumed to be at atmospheric pressure. 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 assigned flow rates are adjusted accordingly.

The 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, the volume and all of the internal surface area of piping runs are input to the deposition model. No credit is taken for deposition of organic iodine. The deposition of aerosols is modeled as described in Appendix A of AEB-98-03 (Reference 10). The deposition of elemental iodine is modeled using the same formulations, but using the deposition velocities obtained from a correlation used in the RADTRAD computer code. Table 8 provides a summary of the main steam line decontamination factors used in the analysis.

The modeling of the steam line deposition and the assigned model parameter values conservatively reduce the amount of deposition credit obtained. The NRC concluded in Reference 5 that main steam line modeling and deposition credit is consistent with the guidance of RG 1.183 (Reference 1). The only change in the MSIV leakage assumptions in the revised analysis from the one approved in Reference 5 is in the total MSIV leakage rate. Increasing the assumed total MSIV leakage from 150 to 250 scfh, with no more than 100 scfh leakage per line, results in the assumed flow distribution in three steam lines. All other parameters and assumptions remain unchanged.

4.6 Leakage from Emergency Core Cooling Systems

Some fission products released from the fuel into the RCS will be carried to the suppression pool via spillage from the postulated pipe break. Since the suppression pool is a source of water for the ECCS and portions of these systems are located outside the primary containment, potential leakage from these systems is evaluated as a radiation exposure pathway. For the purpose of assessing the consequences of leakage from the ECCS it is conservatively assumed 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. The analysis considers the equivalent of 5 gpm unfiltered ECCS leakage starting at the onset of the LOCA. It is assumed 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. There is no change in the evaluation of ECCS leakage from the evaluation approved by the NRC in Reference 5. Key parameters associated with ECCS leakage are included in Table 3.

4.7 Offsite Doses

The maximum 2-hour TEDE to an individual located at the Exclusion Area Boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the Low Population Zone (LPZ) are provided in Table 9. The resulting offsite doses are slightly higher than the ones approved in Reference 5; however, significant margin to the regulatory limits specified in 10 CFR 50.67 is still available.

4.8 Control Room Dose

The control room emergency filtration 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). During normal operations, the maximum unfiltered outside air makeup volume to the control room ventilation system intake is 4000 cfm. Once isolation occurs, a maximum of 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 and test results indicate that less than 100 cfm of unfiltered inleakage would enter the CRE, the revised analysis conservatively assumes that there is 300 cfm of unfiltered inleakage into the control room envelope. In References 6 and 11, Detroit Edison provided the NRC with the requested information in Generic Letter 2003-01, "Control Room Habitability" for Fermi 2. Since tracer gas testing demonstrated that CRE unfiltered inleakage is below 100 cfm, using 300 cfm in the LOCA analysis provides adequate margin to the actual test results.

Radiation shine is not a significant contributor to the total dose; however, it is accounted for in the control room dose evaluation the same way as in the LOCA analysis approved in Reference 5. The ratio of the control room volume to the CRE volume is used to scale the CREF intake flow, the filtered recirculation flow and the unfiltered inleakage to obtain the control room shine volume. Table 10 provides key parameters associated with control room shine volumes. Table 11 provides the results of control room dose evaluation. The resulting total control room dose is lower than the one approved in Reference 5 and provides a larger margin against the regulatory limit specified in 10 CFR 50.67.

4.9 Summary

The revised LOCA dose consequence analysis performed in support of this license amendment request is based on the analysis previously approved by the NRC in Reference 5. The revised analysis increases the combined secondary containment bypass leakage from 5 to 10 percent of the maximum allowable primary containment leakage rate (L_a) and increases the combined MSIV leakage rate from 150 to 250 scfh. Additionally, the revised analysis reduces the control room envelope unfiltered inleakage from 600 to 300 cfm. The results of the analysis indicate that dose consequences at the EAB, LPZ and control room are within the regulatory dose limits specified in 10 CFR 50.67.

5.0 Regulatory Safety Analysis

5.1 No 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 change to Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.11 and SR 3.6.1.3.12 does not involve a significant hazards consideration for the following reasons:

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

   The increase in the allowed secondary containment bypass leakage limit in SR 3.6.1.3.11 and the increase in the total Main Steam Isolation Valve (MSIV) leakage rate limit have been evaluated in a revision to the analysis of the Loss of Coolant Accident (LOCA). Based on the results of the analysis, it has been demonstrated that, with the requested change, the dose consequences of this limiting Design Basis Accident (DBA) 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, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," and Standard Review Plan (SRP) Section 15.0.1. The proposed change also updates the design basis value for the Control Room Envelope (CRE) unfiltered inleakage based on actual test results. This is acceptable because the assumed value in the analysis is more than three times the worst case test value. The proposed change does not affect the normal design or operation of the facility before the accident; rather, it affects leakage limit assumptions that constitute inputs to the evaluation of the consequences. The radiological consequences of the analyzed LOCA have been evaluated using the plant licensing basis for this accident. The results conclude that the control room and offsite doses remain within applicable regulatory limits. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The change in leakage limits 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. As such the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

   This proposed license amendment involves changes in leakage rate limits for the secondary containment bypass leakage and MSIV leakage. The revised leakage rate limits are used in the LOCA radiological analysis in conjunction with the revised CRE unfiltered inleakage limit. The analysis has been performed using conservative methodologies. Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed LOCA event has been carefully selected and margin has been retained to ensure that the analysis adequately bounds postulated event scenario. The dose consequences of this limiting event 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 the revision to the Technical Specification requirements has been analyzed and doses resulting from the pertinent design basis accident have been found to remain within the regulatory limits. The change continues 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. Therefore, the proposed change 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.

5.2 Applicable Regulatory Requirements

The revision to the LOCA radiological dose analysis is performed using the Alternative Source Term methodology which has been established as the licensing basis for this accident. The regulatory requirements, guidance and reviews provided in 10 CFR 50.67, Regulatory Guide 1.183 and Standard Review Plan 15.0.1 are used in the revised analysis. The RADTRAD computer code used to perform the revision of the LOCA analysis has been accepted by the NRC for use in radiological dose analyses. The calculated TEDE doses to the control room, evacuation area boundary and to the low population zone are all below the regulatory dose limits.

Based on the considerations discussed above, it is concluded that, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operating in the proposed manner, (2) activities will be conducted in compliance with NRC regulations, and (3)

the approval and issuance of this proposed amendment will not be inimical to the common defense and security of the health and safety of the public.

6.0 Environmental Considerations

Detroit Edison has reviewed the proposed change against the criteria of 10 CFR 51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor does it significantly change the types or significantly increase the amounts of effluents that may be released offsite. The proposed change does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed change meets the criteria provided in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

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Table 1: Core Inventory

- Based on bounding 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

Note: † Values taken from RADTRAD BWR_DEF.NIF

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	5.0510E+05
Rh-105	2.6240E+04		Pu-238	8.1620E+01
Sb-127	2.2780E+03		Pu-239	1.0410E+01
Sb-129	8.5070E+03		Pu-240	1.8260E+01
Te-127	2.2440E+03		Pu-241	3.8470E+03
Te-127m	3.7990E+02		Am-241	4.9020E+00
Te-129	8.0840E+03		Cm-242	1.2330E+03
Te-129m	1.6390E+03		Cm-244	5.3210E+01

Table 2: Fission Product Release Parameters

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**

Table 3: 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 in each of the two shortest lines – B & D and 50 scfh in the next shortest line - A) Reduced by 50% 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	10.0% 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%
Release Location Primary Containment Leakage Secondary Containment Bypass Leakage MSIV Leakage ECCS	 SGTS Stack TB HVAC Exhaust TB HVAC Exhaust SGTS Stack

Table 4: Main Steam Line Piping Leakage & Deposition Credit		
Parameter	***Value***	***Basis***
1) Leakage Limits	MSIV leakage per line: ≤ 100 scfh Total MSIV leakage for all lines: ≤ 250 scfh	Assumed values. 100 scfh leak is assigned to the two shortest steam lines and the 50 scfh balance to the next shortest. This minimizes credit for settling and deposition by conservatively maximizing the velocity of the leakage inside the piping.
2) Steam Line Nodalization	Each steam line is modeled as two, well-mixed nodes. The first node is defined as the piping from the RPV nozzle to the inboard MSIV. The second node is defined as the piping between the inboard (first) and third MSIVs, which is the boundary of the seismically analyzed piping. A limiting faulted steam line upstream of the inboard MSIV is assumed.	This approach is consistent with the nodalization applied in AEB-98-03 for Perry as well as with the previously approved analysis in license amendment number 160 (Reference 5). The Fermi 2 analysis assumes a worst case single active failure of an MSIV (see below) and assumes a faulted steam line. Piping take-off and corresponding line nodalization summary are provided in Tables 5 and 6 below.
3) Single Failure Assumptions	The outboard MSIV is assumed to be the worst case single failure.	An active failure of an inboard MSIV would result in flow through penetration piping to be at containment conditions, i.e. at a lower volumetric flow than the depressurized flow outside of containment. A failure of the outboard MSIV depressurizes and speeds flow through penetration piping. For conservatism, two nodes are used for both steam lines with assigned MSIV leakage in the analysis.
4) Aerosol Deposition	Aerosol removal in the inboard and outboard piping nodes is evaluated using the formulations of the well-mixed methodology in AEB-98-03, Appendix A. This removal credit is implemented in the RADTRAD calculation in the form of an effective filter credit. No credit is assumed for the inboard node of the faulted line. The settling velocity is assumed as the median velocity based on the Monte Carlo analysis presented in AEB-98-03	This approach is consistent with the recommendations of RG 1.183. Furthermore, it is conservative in that only horizontal piping is credited and the settling area is assumed to be only the bottom half of this piping. AEB-98-03 indicates that, based on the conservatism of the well-mixed assumption, the median velocity is an acceptable value. Table 8 provides a summary of the effective pipe filter efficiencies applied to the MSIV piping leak paths evaluated in the RADTRAD models.
5) Elemental Iodine Deposition	Elemental iodine removal in the inboard and outboard piping nodes is also evaluated using the formulations in AEB-98-03, Appendix A. However,	This approach is consistent with the formulations in AEB-98-03 and the guidance of Reg. Guide 1.183.

Table 4: Main Steam Line Piping Leakage & Deposition Credit		
Parameter	***Value***	***Basis***
	Cline* deposition velocities are used, and with total piping credited, since gravitational settling is not an applicable transport mechanism. No credit is assumed for the inboard node of the faulted line.	
6) MS Line Temperature	For the inboard piping node, steam temperature of 554 degrees Fahrenheit is assumed for the 30-day accident duration. For the outboard piping node, a conservative plant-specific multi-step temperature profile is applied.	The application of a constant temperature for inboard piping is conservative. For the outboard node, the applicable temperature at the beginning of each time interval is held constant for the duration of the interval. Table 7 provides the temperature profile used in the analysis.
7) Organic Iodine Deposition	No organic iodine removal is credited.	Conservative
8) Transit Delay Credit	No plug flow based holdup credit will be taken.	Conservative
9) MSIV Containment Leak Rate	LLRT acceptance criterion is converted to its containment leak rate equivalent by applying pressure and temperature adjustment factors: Pressure: 14.7 / (14.7 + 25) Temperature: (554 + 460)/(68 + 460) After 24 hours, the MSIV leak rate is assumed to decrease by 50%. Where: 14.7 psi is standard atmospheric pressure, 25 psig is MSIV test pressure, 554° F is Accident steam temperature, and 68° F is standard atmospheric temperature	The temperature adjustment is an arbitrary conservatism; however, it provides an allowance for other effects such as a potentially higher volumetric flow rate during the estimated 10 minute post-LOCA period before containment pressure falls below the MSIV test pressure. This assumption has been made in accordance with Reg. Guide 1.183, Appendix A. This is supported by the expected post-accident pressure response of the containment and the conservatism inherent in the application of the full MSIV leak rate during the first 24 hours, as well as the calculation of inboard piping filter credit based on a 30-day post-accident constant pipe wall temperature that corresponds to normal steam line conditions with no credit for the expected post-accident cooldown.
10) Flow Rates Inside Piping	Inboard piping flow rates are the same as established above. Flow rates in outboard piping are "expanded" by not applying pressure adjustment.	Credit is taken for outboard MS pipe cooling based on a conservative application of a plant-specific steam line temperature evaluation.

* J.E.Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report, March 26, 1991 (ADAMS ML003683718) as implemented in NUREG/CR-6604, Supplement 2

Table 5: Main Steam Piping Summary

24.1 Main Steam 26 inch pipe ID (inches)
21.6 Main Steam 24 inch pipe ID (inches)

Horizontal Only

Line A	Line B	Line C	Line D	
1500.4	982.6	1629.1	1487.1	24 inch Outboard Piping, length (inches)
706	**462**	**766**	**700**	24 inch Outboard Piping inside surface area (sq. ft.)
317	**208**	**344**	**314**	24 inch Outboard Piping inside volume (cu. ft.)
479.2	497.6	496.2	481.1	26 inch Outboard Piping, length (inches)
252	**262**	**261**	**253**	26 inch Outboard Piping inside surface area (sq. ft.)
126	**131**	**131**	**127**	26 inch Outboard Piping inside volume (cu. ft.)
958	**724**	**1027**	**953**	**Total Outboard Pipe Surface Area Available (sq. ft.)**
444	**339**	**475**	**441**	**Total Outboard Pipe Volume Available (cu. ft.)**

556.7	715.0	715.0	556.7	26 inch Inboard Piping, length (inches)
293	**376**	**376**	**293**	26 inch Inboard Piping inside surface area (sq. ft.)
147	**189**	**189**	**147**	26 inch Inboard Piping inside volume (cu. ft.)
293	**376**	**376**	**293**	**Total Inboard Pipe Surface Area Available (sq. ft.)**
147	**189**	**189**	**147**	**Total Inboard Pipe Volume Available (cu. ft.)**

Total

Line A	Line B	Line C	Line D	
2005.9	1486.9	2134.6	1993.5	24 inch Outboard Piping, length (inches)
944	**700**	**1004**	**938**	24 inch Outboard Piping inside surface area (sq. ft.)
424	**314**	**451**	**421**	24 inch Outboard Piping inside volume (cu. ft.)
479.2	497.6	496.2	481.1	26 inch Outboard Piping, length (inches)
252	**262**	**261**	**253**	26 inch Outboard Piping inside surface area (sq. ft.)
126	**131**	**131**	**127**	26 inch Outboard Piping inside volume (cu. ft.)
1196	**961**	**1265**	**1191**	**Total Outboard Pipe Surface Area Available (sq. ft.)**
550	**446**	**582**	**548**	**Total Outboard Pipe Volume Available (cu. ft.)**

1210.6	1338.3	1338.3	1210.6	26 inch Inboard Piping, length (inches)
637	**704**	**704**	**637**	26 inch Inboard Piping inside surface area (sq. ft.)
320	**353**	**353**	**320**	26 inch Inboard Piping inside volume (cu. ft.)
637	**704**	**704**	**637**	**Total Inboard Pipe Surface Area Available (sq. ft.)**
320	**353**	**353**	**320**	**Total Inboard Pipe Volume Available (cu. ft.)**

Table 6: Main Steam Line Nodalization for Iodine Deposition

Nodalization (Horizontals)

Line A	Line B	Line C	Line D	
556.7	0	715.0	556.7	Node 1 Length (inches)
293*	0	376	293*	Node 1 Surface Area (sq. ft.)
147	0	189	147	Node 1 Volume (cu. ft.)
1979.6	1480.1	2125.3	1968.2	Node 2 Length (inches)
958*	724*	1027	953*	Node 2 Surface Area (sq. ft.)
444	339	475	441	Node 2 Volume (cu. ft.)

Nodalization (Totals)

Line A	Line B	Line C	Line D	
1210.6	0	1338.3	1210.6	Node 1 Length (inches)
637	0	704	637	Node 1 Surface Area (sq. ft.)
320	0	353	320	Node 1 Volume (cu. ft.)
2485.1	1984.5	2630.8	2474.6	Node 2 Length (inches)
1196	961	1265	1191	Node 2 Surface Area (sq. ft.)
550	446	582	548	Node 2 Volume (cu. ft.)

* Only 50 percent of this surface area is considered as the settling area for aerosol deposition

- No iodine deposition credit is taken for the inboard node of line B (assumed break) and for both nodes of line C (no flow assumed)

Table 7: Outboard MSL Temperature Profile	
Interval	***Temperature (degrees Fahrenheit)***
0-8 hours	554.00
8-16 hours	510.69
16-24 hours	472.08
24-96 hours	437.66
96-720 hours	289.39

Table 8: MSL Decontamination Factors Due to Iodine Deposition

Description	*Inboard A*	*Inboard B*	*Inboard D*	*Outboard A*	*Outboard B*	*Outboard D*
Uncorrected Flow Rate (scfh)	50	100 (faulted)	100	50	100	100
Aerosol Settling Velocity (m/s)	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03
Initial Elemental Deposition Velocity [0-8 hours] (m/s)	5.465E-06	5.465E-06	5.465E-06	5.465E-06	5.465E-06	5.465E-06
Initial Pipe Flow Rate [0-8 hours] corrected for Temp and Press (cfm)	0.593	1.185	1.185	1.600	3.201	3.201
Aerosol Settling Rate Constant (hr^{-1})	1.38E+01	0.00	1.38E+01	1.49E+01	1.48E+01	1.49E+01
Initial Elemental Deposition Rate Constant [0-8 hours] (hr^{-1})	1.29E-01	0.00	1.29E-01	1.40E-01	1.39E-01	1.40E-01
Aerosol Filter Efficiency (0-8 hrs)	**98.27%**	**0.00%**	**96.60%**	**98.57%**	**96.30%**	**97.16%**
Aerosol Filter Efficiency (8-16 hrs)	**98.27%**	**0.00%**	**96.60%**	**98.63%**	**96.45%**	**97.28%**
Aerosol Filter Efficiency (16-24 hrs)	**98.27%**	**0.00%**	**96.60%**	**98.68%**	**96.59%**	**97.39%**
Aerosol Filter Efficiency (24-96 hrs)	**99.13%**	**0.00%**	**98.27%**	**99.36%**	**98.33%**	**98.72%**
Aerosol Filter Efficiency (96-720 hrs)	**99.13%**	**0.00%**	**98.27%**	**99.47%**	**98.60%**	**98.93%**
Elemental Filter Efficiency (0-8 hrs)	**53.61%**	**0.00%**	**36.62%**	**44.56%**	**24.42%**	**28.58%**
Elemental Filter Efficiency (8-16 hrs)	**53.61%**	**0.00%**	**36.62%**	**51.19%**	**29.66%**	**34.31%**
Elemental Filter Efficiency (16-24 hrs)	**53.61%**	**0.00%**	**36.62%**	**57.55%**	**35.27%**	**40.30%**
Elemental Filter Efficiency (24-96 hrs)	**69.80%**	**0.00%**	**53.61%**	**77.61%**	**58.22%**	**63.32%**
Elemental Filter Efficiency (96-720 hrs)	**69.80%**	**0.00%**	**53.61%**	**92.68%**	**83.58%**	**86.32%**

Notes: No flow assumed in line C.
No organic iodine removal is credited.

Table 9: Offsite Radiological Consequence Results			
Location	*Duration*	*TEDE (rem)*	*Regulatory Limit TEDE (rem)*
EAB	Maximum, 2 hours	8.73	25
LPZ	30 days	4.54	25

Table 10: Control Room Shine Volumes	
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 (300 CFM proportioned to the "shine" volume)	67.6 CFM

Table 11: Control Room Radiological Consequence Results	
Dose Contributor	***TEDE Dose (rem)***
Filtered Primary Containment (PC) Leakage (SGTS Filtration Not Credited for First 17 minutes from the Start of the Accident) *[90% of L_A]*	0.49
MSIV Leakage *[250 scfh total all MS lines, 100 scfh max/line]* & Unfiltered PC Leakage Bypassing Secondary Containment *[10% of L_A]*	3.18
ECCS Leakage in Secondary Containment *[5 gpm; 2% Flashing Fraction]*	0.05
Gamma Shine to Control Room (Direct Dose)	0.04
Total Calculated Dose	**3.76**
Regulatory Limit	5.00

References:

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

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

3) NRC Letter to Detroit Edison (Amendment No. 143), "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

4) NRC Letter to Detroit Edison (Amendment No. 144), "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

5) NRC Letter to Detroit Edison (Amendment No. 160), "Fermi 2 – Issuance of Amendment Re: Selective Implementation of Alternative Radiological Source Term Methodology (TAC No. MB7794)," September 28, 2004

6) Detroit Edison Letter to NRC, "Supplemental Response to Generic Letter 2003-01, Control Room Habitability," NRC-05-0042, dated May 25, 2005

7) NRC Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," dated May 2003

8) 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

9) D. A. Powers, K. E. Washington, S. B. Burson, and J. L. Sprung, "A Simplified Model of Aerosol Removal by Natural Process in Reactor Containments," NUREG/CR-6189, SAND94-0407, Sandia National Laboratories, Albuquerque, NM, 1995

10) AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term", USNRC, Accident Evaluation Branch, Devision of Systems Technology, Office of Nuclear Regulatory Research, December 9, 1998

11) Detroit Edison's Letter to NRC, "Detroit Edison's 180-Day Response to Generic Letter 2003-01, Control Room Habitability," NRC-03-0090, dated December 8, 2003

ENCLOSURE 2
TO NRC-05-0045

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

REQUEST TO REVISE THE SECONDARY CONTAINMENT BYPASS LEAKAGE AND MSIV LEAKAGE RATE LIMITS IN SR 3.6.1.3.11 AND SR 3.6.1.3.12

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Affected Page:
3.6-17

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	Verify the combined leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system is ≤ 0.05 L_a when pressurized to ≥ 56.5 psig. (0.05 → 0.10)	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 ≤ 150 scfh and ≤ 100 scfh for any one steam line when tested at ≥ 25 psig. (150 → 250)	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

ENCLOSURE 3
TO NRC-05-0045

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

REQUEST TO REVISE THE SECONDARY CONTAINMENT BYPASS LEAKAGE AND MSIV LEAKAGE RATE LIMITS IN SR 3.6.1.3.11 AND SR 3.6.1.3.12

CLEAN TS PAGE

New Pages:
3.6-17

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	Verify the combined leakage rate for all secondary containment bypass leakage paths that are not provided with a seal system is $\leq$ 0.10 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

ENCLOSURE 4
TO NRC-05-0045

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

REQUEST TO REVISE THE SECONDARY CONTAINMENT BYPASS LEAKAGE AND MSIV LEAKAGE RATE LIMITS IN SR 3.6.1.3.11 AND SR 3.6.1.3.12

MARKED-UP TS BASES PAGES
(For Information Only)

Affected pages:
B 3.6.1.3-16
B 3.6.1.3-17

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

0.10

This SR ensures that the leakage rate of secondary containment bypass leakage paths is $\leq$ 0.05 L_a. 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

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.

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 $\leq$ 150 scfh, and $\leq$ 100 scfh for any one steam line, when tested at $\geq P_t$ (25 psig). 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. 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. 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.

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 ($\geq$ 62.2 psig). Additionally, a combined leakage rate limit of $\leq$ 5 gpm when tested at 1.1 P_a ($\geq$ 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

ENCLOSURE 5
TO NRC-05-0045

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

REQUEST TO REVISE THE SECONDARY CONTAINMENT BYPASS LEAKAGE AND MSIV LEAKAGE RATE LIMITS IN SR 3.6.1.3.11 AND SR 3.6.1.3.12

RADTRAD COMPUTER PROGRAM FILES & NUCLIDE INFORMATION FILES

(For Information Only)

```
Radtrad 3.03 4/15/2001
 3499 MWth Power Level - Filtered PC Leakage Only, 300 cfm CR bypass
 Nuclide Inventory File:
 c:\program files\radtrad3-03\defaults\fermiast-loca (corrected).nif
 Plant Power Level:
  3.4990E+03
 Compartments:
    5
 Compartment 1:
 Containment
    3
  2.9463E+05
    0
    0
    0
    1
    0
 Compartment 2:
 Reactor Building
    3
  1.0000E+00
    0
    0
    0
    0
    0
 Compartment 3:
 Environment
    2
  0.0000E+00
    0
    0
    0
    0
    0
 Compartment 4:
 Control Room
    1
  5.6960E+04
    0
    0
    1
    0
    0
 Compartment 5:
 Hold
    3
  1.0000E+00
    0
    0
    0
    0
    0
 Pathways:
    9
 Pathway 1:
 Containment to Reactor Building
```

```
  1
  2
  4
Pathway 2:
Filtered Environment to Control Room
  3
  4
  2
Pathway 3:
Control Room to  Environment
  4
  3
  2
Pathway 4:
Unfiltered Environment to Control Room
  3
  4
  2
Pathway 5:
MS Line B Containment to Hold
  1
  5
  2
Pathway 6:
MS Line D Containment to Hold
  1
  5
  2
Pathway 7:
MS Line A  Containment to Hold
  1
  5
  2
Pathway 8:
Containment Leakage Bypassing Secondary Containment to Hold
  1
  5
  4
Pathway 9:
Reactor Building to  Environment
  2
  3
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  1
  1   1.0000E+00
c:\program files\radtrad3-03\defaults\fgr11&12.inp
c:\program files\radtrad3-03\defaults\bwr_dba.rft
  0.0000E+00
  1
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
```

```
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  5
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  0
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 4:
  0
  1
  0
  0
  0
  0
  1
  2.7045E+02
  1
  0.0000E+00   9.5000E+01   9.5000E+01   9.5000E+01
  0
  0
Compartment 5:
  0
  1
  0
```

```
  0
  0
  0
  0
  0
  0
Pathways:
  9
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   4.5000E-01
  2.4000E+01   2.2500E-01
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   4.0570E+02   9.9750E+01   9.9750E+01   9.9750E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   4.7330E+02   1.0000E+02   1.0000E+02   1.0000E+02
  5.0000E-01   4.7330E+02   1.0000E+02   1.0000E+02   1.0000E+02
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
```

```
  0
Pathway 4:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.7650E+01   0.0000E+00   0.0000E+00   0.0000E+00
  5.0000E-01   6.7650E+01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 5:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0390E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   1.1850E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 6:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0390E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   1.1850E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 7:
  0
  0
```

```
  0
  0
  0
  1
  3
  0.0000E+00   1.0190E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   2.9630E-01   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 8:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   5.0000E-02
  2.4000E+01   2.5000E-02
  7.2000E+02   0.0000E+00
  0
Pathway 9:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   1.0000E+05   0.0000E+00   0.0000E+00   0.0000E+00
  2.5000E-01   1.0000E+05   9.9000E+01   9.9000E+01   9.9000E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
EAB
  3
  1
  3
  0.0000E+00   0.0000E+00
  1.1000E+00   2.0900E-04
```

```
  3.1000E+00   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.7500E-04
  2.4000E+01   2.3200E-04
  7.2000E+02   0.0000E+00
  0
Location 2:
LPZ
  3
  1
  5
  0.0000E+00   2.1700E-05
  8.0000E+00   1.4500E-05
  2.4000E+01   6.0200E-06
  9.6000E+01   1.7100E-06
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.7500E-04
  2.4000E+01   2.3200E-04
  7.2000E+02   0.0000E+00
  0
Location 3:
Control Room
  4
  0
  1
  2
  0.0000E+00   3.4700E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  6
  0.0000E+00   6.1800E-04
  2.0000E+00   4.5300E-04
  8.0000E+00   1.8800E-04
  2.4000E+01   1.2600E-04
  9.6000E+01   8.7000E-05
  7.2000E+02   0.0000E+00
Simulation Parameters:
  2
  0.0000E+00   1.0000E-01
  1.2000E+01   0.0000E+00
Output Filename:
C:\Documents and Settings\Aleem Boatright\My Documents\My Work\Fermi
2\AST\LOCA Rev 3\RADTRAD\Fermi 2 Filtered PC Leakage - (with 250scfh MSIV
Leak) smallCR-with 300 CR unfiltered.o0
  1
```

```
  2
  1
  0
  1
End of Scenario File
```

```
Radtrad 3.03 4/15/2001
 3499MWth Power Level - Fermi Unit 2 MSIV Leakage (250scfh) and Unfiltered SC
Bypass PC Leakage (10% La) - Separate Compartmentalized Inboard/Outboard
Piping Deposition Credit - AEB-98-03 Type Dep Model - 300 cfm CR Unfiltered
Inleakage
 Nuclide Inventory File:
 c:\program files\radtrad3-03\defaults\fermiast-loca (corrected).nif
 Plant Power Level:
  3.4990E+03
 Compartments:
  10
 Compartment 1:
 Containment
    3
  2.9460E+05
    0
    0
    0
    1
    0
 Compartment 2:
 MS Line B Inboard
    3
  1.0000E-03
    0
    0
    0
    0
    0
 Compartment 3:
 MS Line D Inboard
    3
  3.2000E+02
    0
    0
    0
    0
    0
 Compartment 4:
 MS Line A Inboard
    3
  3.2000E+02
    0
    0
    0
    0
    0
 Compartment 5:
 Environment
    2
  0.0000E+00
    0
    0
    0
    0
    0
 Compartment 6:
```

```
Control Room
  1
 5.6960E+04
  0
  0
  1
  0
  0
Compartment 7:
MS Line B Outboard
  3
 4.4600E+02
  0
  0
  0
  0
  0
Compartment 8:
MS Line D Outboard
  3
 5.4800E+02
  0
  0
  0
  0
  0
Compartment 9:
MS Line A Outboard
  3
 5.5000E+02
  0
  0
  0
  0
  0
Compartment 10:
Hold
  3
 1.0000E+00
  0
  0
  0
  0
  0
Pathways:
 14
Pathway 1:
Containment to Environment (Unfiltered PC Leakage Bypassing SC)
  1
  5
  4
Pathway 2:
Containment Leak to Node 1 MSL B
  1
  2
  2
Pathway 3:
```

```
Containment Leak to Node 1 MSL D
  1
  3
  2
Pathway 4:
Containment Leak to Node 1 MSL A
  1
  4
  2
Pathway 5:
Filtered Environment to Control Room (Intake)
  5
  6
  2
Pathway 6:
Unfiltered Environment to Control Room (Inleakage)
  5
  6
  2
Pathway 7:
Control Room to Environment (Exhaust)
  6
  5
  2
Pathway 8:
MS Line B Node 1 Inboard to MS Line B Node 2 Outboard
  2
  7
  2
Pathway 9:
MS Line D Node 1 Inboard to MS Line D Node 2 Outboard
  3
  8
  2
Pathway 10:
MS Line A Node 1 Inboard to MS Line A Node 2 Outboard
  4
  9
  2
Pathway 11:
MS Line B Node 2 Outboard to Environment
  7
  5
  2
Pathway 12:
MS Line D Node 2 Outboard to Environment
  8
  5
  2
Pathway 13:
MS Line A Node 2 Outboard to Environment
  9
  5
  2
Pathway 14:
Containment to Hold (Filtered PC Leakage)
  1
```

```
 10
  4
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  1
  1   1.0000E+00
c:\program files\radtrad3-03\defaults\fgr11&12.inp
c:\program files\radtrad3-03\defaults\bwr_dba.rft
  0.0000E+00
  1
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
 10
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  3
  3
  1.0000E+01
  1
  1
  0.0000E+00   0.0000E+00
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  0
  1
  0
  0
  0
  0
  0
  0
```

```
  0
Compartment 4:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 5:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 6:
  0
  1
  0
  0
  0
  0
  1
  2.7050E+02
  2
  0.0000E+00   9.5000E+01   9.5000E+01   9.5000E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
Compartment 7:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 8:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 9:
  0
```

```
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 10:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Pathways:
 14
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   5.0000E-02
  2.4000E+01   2.5000E-02
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0390E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   1.1850E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
```

```
  0
  0
  0
  0
  1
  4
  0.0000E+00   2.0385E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   1.1852E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 4:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   1.0193E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   2.9630E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 5:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   4.0570E+02   9.9750E+01   9.9750E+01   9.9750E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 6:
  0
  0
  0
  0
  0
  1
```

```
  2
  0.0000E+00   6.7650E+01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 7:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   4.7325E+02   1.0000E+02   1.0000E+02   1.0000E+02
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 8:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   1.1850E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.4000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  9.6000E+01   5.9260E-01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 9:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   1.1850E+00   9.6600E+01   3.6620E+01   0.0000E+00
  2.4000E+01   5.9260E-01   9.8270E+01   5.3610E+01   0.0000E+00
  9.6000E+01   5.9260E-01   9.8270E+01   5.3610E+01   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
```

```
  0
  0
  0
  0
Pathway 10:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   5.9260E-01   9.8270E+01   5.3610E+01   0.0000E+00
  2.4000E+01   2.9630E-01   9.9130E+01   6.9800E+01   0.0000E+00
  9.6000E+01   2.9630E-01   9.9130E+01   6.9800E+01   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 11:
  0
  0
  0
  0
  0
  1
  6
  0.0000E+00   3.2010E+00   9.6300E+01   2.4420E+01   0.0000E+00
  8.0000E+00   3.0640E+00   9.6450E+01   2.9660E+01   0.0000E+00
  1.6000E+01   2.9420E+00   9.6590E+01   3.5270E+01   0.0000E+00
  2.4000E+01   1.4170E+00   9.8330E+01   5.8220E+01   0.0000E+00
  9.6000E+01   1.1830E+00   9.8600E+01   8.3580E+01   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 12:
  0
  0
  0
  0
  0
  1
  6
  0.0000E+00   3.2008E+00   9.7160E+01   2.8580E+01   0.0000E+00
  8.0000E+00   3.0640E+00   9.7280E+01   3.4310E+01   0.0000E+00
  1.6000E+01   2.9422E+00   9.7390E+01   4.0300E+01   0.0000E+00
  2.4000E+01   1.4168E+00   9.8720E+01   6.3320E+01   0.0000E+00
  9.6000E+01   1.1827E+00   9.8930E+01   8.6320E+01   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
```

```
  0
  0
  0
  0
  0
Pathway 13:
  0
  0
  0
  0
  0
  1
  6
  0.0000E+00   1.6004E+00   9.8570E+01   4.4560E+01   0.0000E+00
  8.0000E+00   1.5320E+00   9.8630E+01   5.1190E+01   0.0000E+00
  1.6000E+01   1.4711E+00   9.8680E+01   5.7550E+01   0.0000E+00
  2.4000E+01   7.0840E-01   9.9360E+01   7.7610E+01   0.0000E+00
  9.6000E+01   5.9140E-01   9.9470E+01   9.2680E+01   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 14:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   4.5000E-02
  2.4000E+01   2.2500E-02
  7.2000E+02   0.0000E+00
  0
Dose Locations:
  3
Location 1:
Control Room
  6
  0
  1
  2
  0.0000E+00   3.4700E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
```

```
  7.2000E+02   0.0000E+00
 Location 2:
 EAB
   5
   1
   3
   0.0000E+00   0.0000E+00
   1.1000E+00   2.0900E-04
   3.1000E+00   0.0000E+00
   1
   4
   0.0000E+00   3.4700E-04
   8.0000E+00   1.7500E-04
   2.4000E+01   2.3200E-04
   7.2000E+02   0.0000E+00
   0
 Location 3:
 LPZ
   5
   1
   5
   0.0000E+00   2.1700E-05
   8.0000E+00   1.4500E-05
   2.4000E+01   6.0200E-06
   9.6000E+01   1.7100E-06
   7.2000E+02   0.0000E+00
   1
   4
   0.0000E+00   3.5000E-04
   8.0000E+00   1.8000E-04
   2.4000E+01   2.3000E-04
   7.2000E+02   0.0000E+00
   0
 Effective Volume Location:
   1
   6
   0.0000E+00   3.1000E-04
   2.0000E+00   2.3300E-04
   8.0000E+00   9.9300E-05
   2.4000E+01   7.0800E-05
   9.6000E+01   5.4800E-05
   7.2000E+02   0.0000E+00
 Simulation Parameters:
   1
   0.0000E+00   0.0000E+00
 Output Filename:
 C:\Documents and Settings\Aleem Boatright\My Documents\My Work\Fermi
2\AST\LOCA Rev 3\RADTRAD\Fermi 2 MSIV and SC Bypass Leakage - 250 scfh Leak -
300cfm CR Unfilt Inleak.o0
   1
   1
   1
   0
   0
 End of Scenario File
```

```
Radtrad 3.03 4/15/2001
 3499 MWth Power Level ECCS Peak w reduced flashing, 15 min SC bypass, 300
cfm unfiltered CR intake
 Nuclide Inventory File:
 c:\program files\radtrad3-03\defaults\fermiast-eccs.nif
 Plant Power Level:
  3.4990E+03
 Compartments:
   4
 Compartment 1:
 ECCS FLUID
   3
  9.4934E+05
   0
   0
   0
   0
   0
 Compartment 2:
 Reactor Building
   3
  1.0000E+00
   0
   0
   0
   0
   0
 Compartment 3:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 4:
 Control Room
   1
  5.6960E+04
   0
   0
   1
   0
   0
 Pathways:
   5
 Pathway 1:
 ECCS FLUID to Reactor Building
   1
   2
   2
 Pathway 2:
 Reactor Building to Environment
   2
   3
   2
```

```
Pathway 3:
Environment to Control Room
  3
  4
  2
Pathway 4:
Control Room to  Environment
  4
  3
  2
Pathway 5:
Unfiltered Environment to Control Room
  3
  4
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  1
  1   1.0000E+00
c:\program files\radtrad3-03\defaults\fgr11&12.inp
c:\program files\radtrad3-03\defaults\bwr_dba.rft
  0.0000E+00
  1
  9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  4
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
```

```
Compartment 3:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 4:
  0
  1
  0
  0
  0
  0
  1
  2.7045E+02
  1
  0.0000E+00   9.5000E+01   9.5000E+01   9.5000E+01
  0
  0
Pathways:
  5
Pathway 1:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   5.0000E+00   9.8000E+01   9.8000E+01   9.8000E+01
  2.4000E+01   5.0000E+00   9.8000E+01   9.8000E+01   9.8000E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   1.0000E+05   0.0000E+00   0.0000E+00   0.0000E+00
  2.5000E-01   1.0000E+05   9.9000E+01   9.9000E+01   9.9000E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
```

```
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   4.0570E+02   9.9750E+01   9.9750E+01   9.9750E+01
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 4:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   4.7325E+02   1.0000E+02   1.0000E+02   1.0000E+02
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 5:
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   6.7650E+01   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
EAB
  3
  1
  3
```

```
  0.0000E+00   0.0000E+00
  1.1000E+00   2.0900E-04
  3.1000E+00   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.7500E-04
  2.4000E+01   2.3200E-04
  7.2000E+02   0.0000E+00
  0
Location 2:
LPZ
  3
  1
  5
  0.0000E+00   2.1700E-05
  8.0000E+00   1.4500E-05
  2.4000E+01   6.0200E-06
  9.6000E+01   1.7100E-06
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   3.4700E-04
  8.0000E+00   1.7500E-04
  2.4000E+01   2.3200E-04
  7.2000E+02   0.0000E+00
  0
Location 3:
Control Room
  4
  0
  1
  2
  0.0000E+00   3.4700E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  6
  0.0000E+00   3.1000E-04
  2.0000E+00   2.3300E-04
  8.0000E+00   9.9300E-05
  2.4000E+01   7.0800E-05
  9.6000E+01   5.4800E-05
  7.2000E+02   0.0000E+00
Simulation Parameters:
  1
  0.0000E+00   0.0000E+00
Output Filename:
C:\Documents and Settings\Aleem Boatright\My Documents\My Work\Fermi
2\AST\LOCA Rev 3\RADTRAD\Fermi 2 ECCS Leakage - 15min drawdown reduced flash
- 300cfm CR Unfilt Inleak.o0
```

```
 1
 2
 1
 0
 1
End of Scenario File
```

FERMIAST-LOCA.NIF

```
Nuclide Inventory Name:  Source Terms per DC-6120, Rev. 0
 FERMI AST LOCA - 35 GWD/MTU 4.58 MW bundle - in Ci/MW
 Power Level:
  0.1000E+01
 Nuclides:
  60
 Nuclide 001:
 Co-58
   7
  0.6117120000E+07
  0.5800E+02
  0.1529E+03
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 002:
 Co-60
   7
  0.1663401096E+09
  0.6000E+02
  0.1830E+03
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 003:
 Kr-85
   1
  0.3382974720E+09
  0.8500E+02
  0.3736E+03
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 004:
 Kr-85m
   1
  0.1612800000E+05
  0.8500E+02
  0.6693E+04
 Kr-85     0.2100E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 005:
 Kr-87
   1
  0.4578000000E+04
  0.8700E+02
  0.1343E+05
 Rb-87     0.1000E+01
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 006:
 Kr-88
   1
  0.1022400000E+05
```

```
 0.8800E+02
 0.1863E+05
Rb-88     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 007:
Rb-86
  3
 0.1612224000E+07
 0.8600E+02
 0.4767E+02
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 008:
Sr-89
  5
 0.4363200000E+07
 0.8900E+02
 0.2609E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 009:
Sr-90
  5
 0.9189573120E+09
 0.9000E+02
 0.3295E+04
Y-90      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 010:
Sr-91
  5
 0.3420000000E+05
 0.9100E+02
 0.3263E+05
Y-91m     0.5800E+00
Y-91      0.4200E+00
none      0.0000E+00
Nuclide 011:
Sr-92
  5
 0.9756000000E+04
 0.9200E+02
 0.3463E+05
Y-92      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 012:
Y-90
  9
 0.2304000000E+06
 0.9000E+02
 0.3405E+04
none      0.0000E+00
```

```
none      0.0000E+00
none      0.0000E+00
Nuclide 013:
Y-91
  9
 0.5055264000E+07
 0.9100E+02
 0.3387E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 014:
Y-92
  9
 0.1274400000E+05
 0.9200E+02
 0.3497E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 015:
Y-93
  9
 0.3636000000E+05
 0.9300E+02
 0.2656E+05
Zr-93     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 016:
Zr-95
  9
 0.5527872000E+07
 0.9500E+02
 0.4575E+05
Nb-95m    0.7000E-02
Nb-95     0.9900E+00
none      0.0000E+00
Nuclide 017:
Zr-97
  9
 0.6084000000E+05
 0.9700E+02
 0.4322E+05
Nb-97m    0.9500E+00
Nb-97     0.5300E-01
none      0.0000E+00
Nuclide 018:
Nb-95
  9
 0.3036960000E+07
 0.9500E+02
 0.4609E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 019:
```

```
Mo-99
  7
 0.2376000000E+06
 0.9900E+02
 0.4988E+05
Tc-99m    0.8800E+00
Tc-99     0.1200E+00
none      0.0000E+00
Nuclide 020:
Tc-99m
  7
 0.2167200000E+05
 0.9900E+02
 0.4428E+05
Tc-99     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 021:
Ru-103
  7
 0.3393792000E+07
 0.1030E+03
 0.4183E+05
Rh-103m   0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 022:
Ru-105
  7
 0.1598400000E+05
 0.1050E+03
 0.2826E+05
Rh-105    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 023:
Ru-106
  7
 0.3181248000E+08
 0.1060E+03
 0.1558E+05
Rh-106    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 024:
Rh-105
  7
 0.1272960000E+06
 0.1050E+03
 0.2624E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 025:
Sb-127
  4
 0.3326400000E+06
```

```
 0.1270E+03
 0.2278E+04
Te-127m  0.1800E+00
Te-127   0.8200E+00
none     0.0000E+00
Nuclide 026:
Sb-129
  4
 0.1555200000E+05
 0.1290E+03
 0.8507E+04
Te-129m  0.2200E+00
Te-129   0.7700E+00
none     0.0000E+00
Nuclide 027:
Te-127
  4
 0.3366000000E+05
 0.1270E+03
 0.2244E+04
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 028:
Te-127m
  4
 0.9417600000E+07
 0.1270E+03
 0.3799E+03
Te-127   0.9800E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 029:
Te-129
  4
 0.4176000000E+04
 0.1290E+03
 0.8084E+04
I-129    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 030:
Te-129m
  4
 0.2903040000E+07
 0.1290E+03
 0.1639E+04
Te-129   0.6500E+00
I-129    0.3500E+00
none     0.0000E+00
Nuclide 031:
Te-131m
  4
 0.1080000000E+06
 0.1310E+03
 0.5246E+04
Te-131   0.2200E+00
```

```
I-131     0.7800E+00
none      0.0000E+00
Nuclide 032:
Te-132
  4
 0.2815200000E+06
 0.1320E+03
 0.3823E+05
I-132     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 033:
I-131
  2
 0.6946560000E+06
 0.1310E+03
 0.2657E+05
Xe-131m   0.1100E-01
none      0.0000E+00
none      0.0000E+00
Nuclide 034:
I-132
  2
 0.8280000000E+04
 0.1320E+03
 0.3901E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 035:
I-133
  2
 0.7488000000E+05
 0.1330E+03
 0.5500E+05
Xe-133m   0.2900E-01
Xe-133    0.9700E+00
none      0.0000E+00
Nuclide 036:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 0.6078E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 037:
I-135
  2
 0.2379600000E+05
 0.1350E+03
 0.5235E+05
Xe-135m   0.1500E+00
Xe-135    0.8500E+00
none      0.0000E+00
Nuclide 038:
```

```
Xe-133
  1
 0.4531680000E+06
 0.1330E+03
 0.5412E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 039:
Xe-135
  1
 0.3272400000E+05
 0.1350E+03
 0.1451E+05
Cs-135    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 040:
Cs-134
  3
 0.6507177120E+08
 0.1340E+03
 0.4793E+04
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 041:
Cs-136
  3
 0.1131840000E+07
 0.1360E+03
 0.1463E+04
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 042:
Cs-137
  3
 0.9467280000E+09
 0.1370E+03
 0.4270E+04
Ba-137m   0.9500E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 043:
Ba-139
  6
 0.4962000000E+04
 0.1390E+03
 0.4843E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 044:
Ba-140
  6
 0.1100736000E+07
```

```
 0.1400E+03
 0.4877E+05
La-140    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 045:
La-140
  9
 0.1449792000E+06
 0.1400E+03
 0.5079E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 046:
La-141
  9
 0.1414800000E+05
 0.1410E+03
 0.4422E+05
Ce-141    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 047:
La-142
  9
 0.5550000000E+04
 0.1420E+03
 0.4320E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 048:
Ce-141
  8
 0.2808086400E+07
 0.1410E+03
 0.4477E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 049:
Ce-143
  8
 0.1188000000E+06
 0.1430E+03
 0.4142E+05
Pr-143    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 050:
Ce-144
  8
 0.2456352000E+08
 0.1440E+03
 0.3790E+05
Pr-144m   0.1800E-01
```

```
Pr-144    0.9800E+00
none      0.0000E+00
Nuclide 051:
Pr-143
  9
 0.1171584000E+07
 0.1430E+03
 0.4041E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 052:
Nd-147
  9
 0.9486720000E+06
 0.1470E+03
 0.1800E+05
Pm-147    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 053:
Np-239
  8
 0.2034720000E+06
 0.2390E+03
 0.5051E+06
Pu-239    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 054:
Pu-238
  8
 0.2768863824E+10
 0.2380E+03
 0.8162E+02
U-234     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 055:
Pu-239
  8
 0.7594336440E+12
 0.2390E+03
 0.1041E+02
U-235     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 056:
Pu-240
  8
 0.2062920312E+12
 0.2400E+03
 0.1826E+02
U-236     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 057:
```

```
Pu-241
  8
 0.4544294400E+09
 0.2410E+03
 0.3847E+04
U-237     0.2400E-04
Am-241    0.1000E+01
none      0.0000E+00
Nuclide 058:
Am-241
  9
 0.1363919472E+11
 0.2410E+03
 0.4902E+01
Np-237    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 059:
Cm-242
  9
 0.1406592000E+08
 0.2420E+03
 0.1233E+04
Pu-238    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 060:
Cm-244
  9
 0.5715081360E+09
 0.2440E+03
 0.5321E+02
Pu-240    0.1000E+01
none      0.0000E+00
none      0.0000E+00
End of Nuclear Inventory File
```

```
Nuclide Inventory Name:  FERMI-ECCS
 FERMI AST ECCS - 35 GWD/MTU 4.58 MW bundle - in Ci/MW
 Power Level:
  0.1000E+01
 Nuclides:
  60
 Nuclide 001:
 Co-58
   7
  0.6117120000E+07
  0.5800E+02
  0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 002:
 Co-60
   7
  0.1663401096E+09
  0.6000E+02
  0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 003:
 Kr-85
   1
  0.3382974720E+09
  0.8500E+02
  0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 004:
 Kr-85m
   1
  0.1612800000E+05
  0.8500E+02
  0.0000E+00
 Kr-85     0.2100E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 005:
 Kr-87
   1
  0.4578000000E+04
  0.8700E+02
  0.0000E+00
 Rb-87     0.1000E+01
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 006:
 Kr-88
   1
  0.1022400000E+05
```

```
 0.8800E+02
 0.0000E+00
Rb-88     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 007:
Rb-86
  3
 0.1612224000E+07
 0.8600E+02
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 008:
Sr-89
  5
 0.4363200000E+07
 0.8900E+02
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 009:
Sr-90
  5
 0.9189573120E+09
 0.9000E+02
 0.0000E+00
Y-90      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 010:
Sr-91
  5
 0.3420000000E+05
 0.9100E+02
 0.0000E+00
Y-91m     0.5800E+00
Y-91      0.4200E+00
none      0.0000E+00
Nuclide 011:
Sr-92
  5
 0.9756000000E+04
 0.9200E+02
 0.0000E+00
Y-92      0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 012:
Y-90
  9
 0.2304000000E+06
 0.9000E+02
 0.0000E+00
none      0.0000E+00
```

```
none      0.0000E+00
none      0.0000E+00
Nuclide 013:
Y-91
  9
 0.5055264000E+07
 0.9100E+02
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 014:
Y-92
  9
 0.1274400000E+05
 0.9200E+02
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 015:
Y-93
  9
 0.3636000000E+05
 0.9300E+02
 0.0000E+00
Zr-93     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 016:
Zr-95
  9
 0.5527872000E+07
 0.9500E+02
 0.0000E+00
Nb-95m    0.7000E-02
Nb-95     0.9900E+00
none      0.0000E+00
Nuclide 017:
Zr-97
  9
 0.6084000000E+05
 0.9700E+02
 0.0000E+00
Nb-97m    0.9500E+00
Nb-97     0.5300E-01
none      0.0000E+00
Nuclide 018:
Nb-95
  9
 0.3036960000E+07
 0.9500E+02
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 019:
```

```
Mo-99
  7
 0.2376000000E+06
 0.9900E+02
 0.0000E+00
Tc-99m    0.8800E+00
Tc-99     0.1200E+00
none      0.0000E+00
Nuclide 020:
Tc-99m
  7
 0.2167200000E+05
 0.9900E+02
 0.0000E+00
Tc-99     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 021:
Ru-103
  7
 0.3393792000E+07
 0.1030E+03
 0.0000E+00
Rh-103m   0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 022:
Ru-105
  7
 0.1598400000E+05
 0.1050E+03
 0.0000E+00
Rh-105    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 023:
Ru-106
  7
 0.3181248000E+08
 0.1060E+03
 0.0000E+00
Rh-106    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 024:
Rh-105
  7
 0.1272960000E+06
 0.1050E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 025:
Sb-127
  4
 0.3326400000E+06
```

```
 0.1270E+03
 0.0000E+00
Te-127m  0.1800E+00
Te-127   0.8200E+00
none     0.0000E+00
Nuclide 026:
Sb-129
  4
 0.1555200000E+05
 0.1290E+03
 0.0000E+00
Te-129m  0.2200E+00
Te-129   0.7700E+00
none     0.0000E+00
Nuclide 027:
Te-127
  4
 0.3366000000E+05
 0.1270E+03
 0.0000E+00
none     0.0000E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 028:
Te-127m
  4
 0.9417600000E+07
 0.1270E+03
 0.0000E+00
Te-127   0.9800E+00
none     0.0000E+00
none     0.0000E+00
Nuclide 029:
Te-129
  4
 0.4176000000E+04
 0.1290E+03
 0.0000E+00
I-129    0.1000E+01
none     0.0000E+00
none     0.0000E+00
Nuclide 030:
Te-129m
  4
 0.2903040000E+07
 0.1290E+03
 0.0000E+00
Te-129   0.6500E+00
I-129    0.3500E+00
none     0.0000E+00
Nuclide 031:
Te-131m
  4
 0.1080000000E+06
 0.1310E+03
 0.0000E+00
Te-131   0.2200E+00
```

```
I-131     0.7800E+00
none      0.0000E+00
Nuclide 032:
Te-132
  4
 0.2815200000E+06
 0.1320E+03
 0.0000E+00
I-132     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 033:
I-131
  2
 0.6946560000E+06
 0.1310E+03
 0.2657E+05
Xe-131m   0.1100E-01
none      0.0000E+00
none      0.0000E+00
Nuclide 034:
I-132
  2
 0.8280000000E+04
 0.1320E+03
 0.3901E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 035:
I-133
  2
 0.7488000000E+05
 0.1330E+03
 0.5500E+05
Xe-133m   0.2900E-01
Xe-133    0.9700E+00
none      0.0000E+00
Nuclide 036:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 0.6078E+05
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 037:
I-135
  2
 0.2379600000E+05
 0.1350E+03
 0.5235E+05
Xe-135m   0.1500E+00
Xe-135    0.8500E+00
none      0.0000E+00
Nuclide 038:
```

```
Xe-133
  1
 0.4531680000E+06
 0.1330E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 039:
Xe-135
  1
 0.3272400000E+05
 0.1350E+03
 0.0000E+00
Cs-135    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 040:
Cs-134
  3
 0.6507177120E+08
 0.1340E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 041:
Cs-136
  3
 0.1131840000E+07
 0.1360E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 042:
Cs-137
  3
 0.9467280000E+09
 0.1370E+03
 0.0000E+00
Ba-137m   0.9500E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 043:
Ba-139
  6
 0.4962000000E+04
 0.1390E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 044:
Ba-140
  6
 0.1100736000E+07
```

```
 0.1400E+03
 0.0000E+00
La-140    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 045:
La-140
  9
 0.1449792000E+06
 0.1400E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 046:
La-141
  9
 0.1414800000E+05
 0.1410E+03
 0.0000E+00
Ce-141    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 047:
La-142
  9
 0.5550000000E+04
 0.1420E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 048:
Ce-141
  8
 0.2808086400E+07
 0.1410E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 049:
Ce-143
  8
 0.1188000000E+06
 0.1430E+03
 0.0000E+00
Pr-143    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 050:
Ce-144
  8
 0.2456352000E+08
 0.1440E+03
 0.0000E+00
Pr-144m   0.1800E-01
```

```
Pr-144    0.9800E+00
none      0.0000E+00
Nuclide 051:
Pr-143
  9
 0.1171584000E+07
 0.1430E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 052:
Nd-147
  9
 0.9486720000E+06
 0.1470E+03
 0.0000E+00
Pm-147    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 053:
Np-239
  8
 0.2034720000E+06
 0.2390E+03
 0.0000E+00
Pu-239    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 054:
Pu-238
  8
 0.2768863824E+10
 0.2380E+03
 0.0000E+00
U-234     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 055:
Pu-239
  8
 0.7594336440E+12
 0.2390E+03
 0.0000E+00
U-235     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 056:
Pu-240
  8
 0.2062920312E+12
 0.2400E+03
 0.0000E+00
U-236     0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 057:
```

```
Pu-241
  8
 0.4544294400E+09
 0.2410E+03
 0.0000E+00
U-237     0.2400E-04
Am-241    0.1000E+01
none      0.0000E+00
Nuclide 058:
Am-241
  9
 0.1363919472E+11
 0.2410E+03
 0.0000E+00
Np-237    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 059:
Cm-242
  9
 0.1406592000E+08
 0.2420E+03
 0.0000E+00
Pu-238    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 060:
Cm-244
  9
 0.5715081360E+09
 0.2440E+03
 0.0000E+00
Pu-240    0.1000E+01
none      0.0000E+00
none      0.0000E+00
End of Nuclear Inventory File
```