William T. O'Connor, Jr.
Vice President, Nuclear Generation

Fermi 2
6400 North Dixie Hwy., Newport, Michigan 48166
Tel: 734.586.5201 Fax: 734.586.4172

Detroit Edison
A DTE Energy Company

10CFR50.67
10CFR50.90

November 21, 2000
NRC-00-0066

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

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

2) U.S. Nuclear Regulatory Commission, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995

3) U. S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962

4) General Electric Nuclear Energy, NEDC-32963A, "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR," March 2000

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

Subject: Proposed License Amendment for a Limited Scope Implementation of the Alternative Source Term Insights in NUREG-1465 Related to the Timing of the Onset of Gap Activity Release

Pursuant to 10 CFR 50.67 and 50.90, Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License NPF-43 to implement one of the insights established in Reference 2 associated with the release of fission products following

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an accident. The proposed licensing basis change takes credit for the time delay in fuel rod gap activity release following a postulated design-basis accident. This change will modify the Fermi 2 Loss of Coolant Accident (LOCA) analysis scenario to recognize that the initial release will consist of reactor coolant activity only. The timing for fission product releases resulting from a perforated fuel rod is delayed by 121 seconds (Reference 5 rounds this value to 2 minutes) instead of the instantaneous release assumed in Reference 3.

Reference 4 provides the technical basis for the requested change. This report presents a conservative estimate of the time delay for the fuel rod gap activity release phase of a design basis accident, for the most limiting Boiling Water Reactor (BWR). The NRC, in a letter to Grand Gulf Nuclear Station, Unit 1, (Docket No. 50-416), dated September 9, 1999, accepted this report for referencing in license amendment applications for all operating BWRs.

Enclosure 1 provides a description and an evaluation of the proposed license change. Enclosure 2 provides an analysis of significant hazards consideration using the standards of 10 CFR 50.92. Enclosure 3 provides marked up sections of the current Updated Final Safety Analysis Report (UFSAR) to show the proposed primary changes and a typed version of the affected UFSAR Sections with the proposed changes incorporated.

Detroit Edison has reviewed the proposed license change against the criteria of 10 CFR 51.22 for categorical exclusion of environmental review. 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 change does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed license 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.

Detroit Edison requests that the NRC approve and issue a license amendment for the proposed changes by July 31, 2001, with an implementation period of within 60 days following NRC approval.

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

Sincerely,

William J. O'Connor Jr.

Enclosures

cc: D. S. Hood
M. A. Ring
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 O'Connor Jr.

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

On this 21 st. day of November, 2000 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.

Notary Public

NORMAN K. PETERSON
Notary Public, Monroe County, MI
My Commission Expires July 24, 2002

ENCLOSURE 1 TO
NRC-00-0066

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

REQUEST FOR LICENSE AMENDMENT:

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATIVE SOURCE TERM INSIGHTS RELATED TO THE ONSET OF GAP ACTIVITY RELEASE TIMING

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGES

DESCRIPTION

Fermi 2 Updated Final Safety Analysis Report (UFSAR), Section 15.6.5, currently includes an analysis of a Loss of Coolant Accident (LOCA) scenario inside the primary containment in which a complete circumferential break of one of the two recirculation loop pipe lines occurs. The analysis assumes an instantaneous release of fission products resulting from a perforated fuel rod into the primary containment. This assumption was based on the Atomic Energy Commission document number TID-14844 (Reference 3), published in 1962. This document established conservative accident source terms for use in the evaluation of the dose consequences of design basis accidents. While the TID-14844 standard is referenced in 10 CFR Part 100, it is as a guide.

To limit the uncontrolled release of radioactive materials to the environs, a timely isolation of penetrations through the primary containment is initiated following an accident. Automatic Primary Containment Isolation Valves (PCIVs) are designed to close with sufficient rapidity to prevent the release of significant amounts of radioactive material from the primary containment. The maximum isolation times for PCIVs are currently specified in the Fermi 2 Technical Requirements Manual (TRM).

Research and operating experience over the past 30 years have expanded the understanding of severe accidents and the behavior of fission product release. NUREG-1465 (Reference 2) was published in 1995 with revised accident source terms for use in the licensing of future Light Water Reactors (LWRs). The NRC, in 10 CFR 50.67, later allowed the use of the Alternative Source Terms (AST) described in NUREG-1465 at operating plants. This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

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

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

Phases 1, 2, and 3 are considered in current design basis accident (DBA) evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST, the coolant activity release is assumed to occur instantaneously and end with the onset of the gap activity release.

The requested license amendment involves a limited-scope application of the AST, addressing the timing and duration of the coolant activity release phase and the timing of the gap activity release phase. This proposed change establishes and documents a quantifiable design basis for the PCIV closure times credited for limiting post-accident doses to both control room personnel and to offsite individuals. Other PCIVs, for which the maximum closure time is based on other functional performance requirements such as line break isolation, are not affected by this change.

NUREG-1465 assumes a coolant activity release phase duration of 30 seconds based solely on estimates for Pressurized Water Reactor (PWR) plants. For Boiling Water Reactor (BWR) plants, the NUREG acknowledges that the coolant activity release phase would last longer and that more specific analyses could justify a longer duration. The BWR Owners Group (BWROG) has performed a conservative analysis (Reference 4) to determine the minimum time to gap activity release for a generic BWR following a DBA LOCA with no emergency core cooling system (ECCS) injection. The analysis includes sensitivity studies to determine the most limiting BWR design, fuel type, and core burnup. It shows that a BWR-4 plant, with a reactor pressure vessel (RPV) inside diameter (ID) of 205 inches, 28-inch ID recirculation pipes, and GE-11 (9X9 lattice) fuel configuration is the most limiting design. The NRC, in a letter dated September 9, 1999 to Grand Gulf Nuclear Station, approved the BWROG analysis for use by all BWRs in support of plant-specific license amendments.

Reference 4 documents the results of an analysis performed to determine the minimum time to the onset of release of radioactive material from a perforated fuel assembly following a DBA LOCA at a generic BWR. The parameters used in the analysis match those for Fermi 2 except for the more limiting RPV ID of 205 inches as compared to 251 inches for Fermi 2. NRC-approved computer codes were used to calculate the minimum duration of the coolant activity release phase described in NUREG-1465. The BWR coolant activity release phase, which represents the period from the time of the start of the accident until the initiation of a perforated fuel gap activity release, is calculated to last 121 seconds. Regulatory Guide 1.183 rounds this value to 2 minutes. The values requested in this change to the licensing basis are derived from the conclusions reached in Reference 4.

EVALUATION OF THE PROPOSED CHANGES

Based on the results of NUREG-1465 and on the BWR-specific value for the timing of the gap activity release phase for a LOCA, as calculated in Reference 4, it is proposed that the Fermi 2 licensing basis be changed from the TID-14844 assumption of an instantaneous release of fission products into containment following a LOCA, to the more realistic assumption that the gap activity release is delayed by 121 seconds.

Detroit Edison has performed a site-specific analysis of a release of radioactivity from the reactor coolant during the first 121 seconds of an accident. The analysis assumes no containment isolation, no secondary containment holdup, and no emergency ventilation system filtration during this time; however, it takes credit for iodine scrubbing in the suppression pool in accordance with NUREG-0800, Section 6.5.5. Assuming the maximum coolant iodine activity permitted by the Technical Specifications, the 2-hour Exclusion Area Boundary (EAB) dose associated with this release is conservatively determined to be less than 2 rem thyroid. This dose is insignificant with respect to the thyroid dose shown in UFSAR Table 15.6.5-4 and the acceptance limit (300 rem) contained in 10 CFR Part 100.

The only change proposed here is the one involving the timing and duration of fission product releases described in NUREG-1465. Other insights involving the composition and magnitude of the release, the chemical form, and the removal mechanisms are not included in this request. Detroit Edison plans to submit a separate license amendment request in the near future to utilize the AST in the reanalysis of the Fuel Handling Accident.

The proposed change does not affect the composition, magnitude or chemical form of the accident source term. The source term defined by TID-14844 is still utilized in the design basis analysis. The only change involves the delay in the gap activity release; therefore, all other assumptions used in the current licensing basis remain unchanged. The delay in fission product release resulting from a fuel rod perforation does not affect the dose consequences of the LOCA analysis because the same amount of radioactivity is assumed to be released.

The primary change to the Fermi 2 UFSAR to reflect this change in the licensing basis will be a revision to Sections 15.6.5.5.1, 15.6.5.5.2, and 15.6.5.5.3. A markup of these sections showing this revision is included in Enclosure 3. Once the new licensing basis has been approved by the NRC, the maximum isolation time for selected primary containment isolation valves (PCIVs) would potentially be increased to 108 seconds or less. This value is derived from UFSAR Table 8.3-5 for the case of simultaneous LOCA with a loss of offsite power. The 108-second maximum isolation time allows for 3 seconds from the occurrence of the accident before the Emergency Diesel Generators (EDGs) start, and another 10 seconds before the EDG breakers close. This ensures that the PCIVs are closed and the containment is isolated within 121 seconds following the accident and prior to any expected release of fission products from damaged fuel. Table 1 provides a list of valves that may be affected by this change. This Table includes several

Air Operated Valves (AOVs) for which EDG power is not required; however, the 108-second potential maximum isolation time is conservatively used.

The actual change to selected PCIV closure times would be handled in accordance with the 10 CFR 50.59 program. The allowable automatic PCIV closure times are currently presented in the Fermi 2 Technical Requirements Manual, Table TR3.6.3-1. Nominal values are also shown in Table 6.2-2 of the UFSAR. The 10 CFR 50.59 process will address any potential concerns regarding the longer closure times of the specific valves involved. Other pertinent revisions to the UFSAR would also be done at the time of PCIV closure time changes.

Table 1 includes selected PCIVs that are designed to provide containment isolation to restrict the release of radioactive material to the environs. Other PCIVs have closure times reflecting system performance requirements, equipment qualification, high energy line break mitigation, or other regulatory requirements. These valves are not included in Table 1 and are not affected by the proposed change.

The maximum closure times for valves included in Table 1 are currently in accordance with regulatory guidelines in Standard Review Plan (SRP) 6.2.4. This document recommends a general PCIV closure time of 60 seconds or less. This guidance resulted from the extremely conservative assumptions in the TID-14844 standard and was intended to limit radiological consequences following a LOCA to within the limits contained in 10 CFR 100. For containment purge and vent isolation valves, a 5-second closure time was recommended because these valves provide an open path from the containment to the environment. However, using the insights contained in NUREG-1465, Detroit Edison has performed a site-specific analysis and has concluded that the dose consequences of longer closure times for these valves are in conformance with 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criteria 19 guidelines.

The proposed change in the timing of the fission products release following a LOCA has no direct effect on the probability of the accident; therefore, it will not impact the core damage frequency (CDF) or the large early release frequency (LERF) of the Fermi 2 probabilistic safety assessment (PSA).

Table 1
Automatic Primary Containment Isolation Valves
Potentially Impacted

Valve	Description	Current Maximum Isolation Time (Seconds)	Potential Maximum Isolation Time (Seconds)
B3100-F014A	Recirculation Pump Seal Purge Isolation Valve	5	108
B3100-F014B	Recirculation Pump Seal Purge Isolation Valve	5	108
B3100-F016A	Recirculation Pump Seal Purge Isolation Valve	5	108
B3100-F016B	Recirculation Pump Seal Purge Isolation Valve	5	108
E1150-F021A	Residual Heat Removal (RHR) Drywell Spray Isolation Valve	60	108
E1150-F021B	RHR Drywell Spray Isolation Valve	60	108
E1150-F024A	RHR Containment Cooling/Test Isolation Valve	60	108
E1150-F024B	RHR Containment Cooling/Test Isolation Valve	60	108
E1150-F027A	RHR Suppression Pool Spray Isolation Valve	60	108
E1150-F027B	RHR Suppression Pool Spray Isolation Valve	60	108
E1150-F028A	RHR Suppression Pool Spray/Test Isolation Valve	60	108
E1150-F028B	RHR Suppression Pool Spray/Test Isolation Valve	60	108
E4150-F042	High Pressure Coolant Injection (HPCI) Booster Pump Suction from Suppression Chamber Isolation Valve	60	108
E4150-F075	HPCI Turbine Exhaust Line Vacuum Breaker Isolation Valve	60	108
E4150-F079	HPCI Turbine Exhaust Line Vacuum Breaker Isolation Valve	60	108
E5150-F062	Reactor Core Isolation Cooling (RCIC) Turbine Exhaust Line Vacuum Breaker Isolation Valve	60	108
E5150-F084	RCIC Turbine Exhaust Line Vacuum Breaker Isolation Valve	60	108
G1154-F600	Drywell Floor Drain Sump Pump Discharge Isolation Valve	60	108
G1100-F003	Drywell Floor Drain Sump Pump Discharge Isolation Valve	60	108
G1154-F018	Drywell Equipment Drain Sump Pump Discharge Isolation Valve	60	108
G1100-F019	Drywell Equipment Drain Sump Pump Discharge Isolation Valve	60	108
G5100-F600	Torus Water Management System (TWMS) Torus Drain Isolation Valve	60	108
G5100-F601	TWMS Torus Drain Isolation Valve	60	108
G5100-F602	TWMS Torus Drain Isolation Valve	60	108
G5100-F603	TWMS Torus Drain Isolation Valve	60	108
G5100-F604	TWMS to RHR Line Isolation Valve	60	108
G5100-F605	TWMS to RHR Line Isolation Valve	60	108
G5100-F606	TWMS to CSS Test Line Isolation Valve	60	108
G5100-F607	TWMS to CSS Test Line Isolation Valve	60	108

Table 1
Automatic Primary Containment Isolation Valves
Potentially Impacted
(continued)

Valve	Description	Current Maximum Isolation Time (Seconds)	Potential Maximum Isolation Time (Seconds)
T4600-F400	Suppression Pool Exhaust Air Purge to Standby Gas Treatment System (SGTS) Isolation Valve	5	108
T4600-F401	Suppression Pool Exhaust Air Purge to SGTS Isolation Valve	5	108
T4600-F402	Drywell Exhaust to SGTS Isolation Valve	5	108
T4600-F411	Drywell Exhaust to SGTS Bypass Isolation Valve	5	108
T4600-F412	Suppression Pool Exhaust Air Purge to SGTS Bypass Isolation Valve	5	108
T4800-F404	Suppression Pool Nitrogen Inlet Isolation Valve	5	108
T4800-F405	Suppression Pool Vent Valve	5	108
T4800-F407	Drywell Air Purge Inlet Vent Valve	5	108
T4800-F408	Drywell Nitrogen Inlet Isolation Valve	5	108
T4800-F409	Suppression Pool Nitrogen Inlet Isolation Valve	5	108
T4800-F410	Suppression Pool Nitrogen Inlet to SGTS Isolation Valve	5	108
T4800-F453	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4800-F454	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4800-F455	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4800-F456	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4800-F457	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4800-F458	Containment Nitrogen Pressure Control Isolation Valve	60	108
T4803-F601	Drywell Air Purge Inlet Isolation Valve	5	108
T4803-F602	Drywell Exhaust Isolation Valve	5	108
T4901-F601	Nitrogen Inlet to Drywell Inboard Isolation Valve	60	108
T4901-F602	Nitrogen Inlet to Drywell Inboard Isolation Valve	60	108
T4901-F465	Nitrogen Inlet to Drywell Outboard Isolation Valve	60	108
T4901-F468	Nitrogen Inlet to Drywell Outboard Isolation Valve	60	108
T5000-F450	Primary Containment Radiation Monitoring Isolation Valve	60	108
T5000-F451	Primary Containment Radiation Monitoring Isolation Valve	60	108
T5000-F455	Primary Containment Radiation Monitoring Isolation Valve	60	108
T5000-F456	Primary Containment Radiation Monitoring Isolation Valve	60	108

ENCLOSURE 2 TO
NRC-00-0066

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

REQUEST FOR LICENSE AMENDMENT:

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATIVE SOURCE TERM INSIGHTS RELATED TO THE ONSET OF GAP ACTIVITY RELEASE TIMING

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

In accordance with 10CFR50.92, Detroit Edison has made a determination that the proposed amendment involves no significant hazards consideration. The License Amendment described above does not involve a significant hazards consideration for the following reasons:

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

   The proposed change takes credit for one of the alternative source term (AST) insights contained in NUREG-1465 which recognizes that fission product release from a fuel assembly is not instantaneous in a design basis accident. Implementation of this change into the licensing basis will be used to justify an increase in the maximum allowable closure times for primary containment isolation valves. A change in the timing of the gap release does not affect the precursors for any accident or transient previously evaluated as part of the Fermi 2 licensing basis. Therefore, there is no increase in the probability of any accident.

   A plant specific radiological analysis has been performed to evaluate the effects of extending the maximum allowable valve closure times on accident dose consequences. This evaluation utilized the insights contained in NUREG-1465 (Reference 2) and NEDC-32963A (Reference 4) to justify no gap activity release during the initial 121 seconds of the accident. Therefore, during this period, the only releases are from reactor coolant activity. Assuming the maximum coolant iodine activity permitted in the Technical Specifications, the 2-hour Exclusion Area Boundary (EAB) dose associated with this release has been conservatively estimated to be less than 2 rem thyroid. This dose represents a small fraction of the LOCA dose evaluated in the UFSAR and is significantly lower than the 300 rem thyroid dose acceptance limit in 10 CFR Part 100.

   UFSAR Figures 6.2-9 and 6.2-11 show the DBA LOCA primary containment pressure response. These figures indicate that drywell pressure peaks at around 5 seconds into the accident before gradually dropping off; therefore, PCIVs would not be required to close against increased containment pressure as a result of this change.

   Utilizing all of the insights contained in NUREG-1465, would result in a reduction in the calculated dose. However, because this request is for a selective implementation of the AST scope, crediting only the timing of the gap activity release, the long term dose calculations based on TID-14844 in the UFSAR are not changed. Therefore, it is concluded that the proposed change does not significantly increase the consequences of a previously evaluated accident.

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

   The primary containment isolation system is designed to prevent the unfiltered release of radioactive material to the environs following an accident. Therefore, the system is relied upon to mitigate the dose consequences of an accident. The proposed change recognizes the time delay before fission products are released into the containment as a result of fuel damage and allows for the adjustment of the maximum PCIV closure times accordingly. This change does not affect the function of the primary containment isolation system. The relaxation in valve closure times will be applied only to valves that do not have other system performance requirements on isolation time. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

   The proposed change revises the Fermi 2 licensing basis for the offsite dose calculations during the initial 121 seconds of a LOCA scenario. For this period of time, only coolant activity release is postulated. No fission product release from perforated fuel rods is assumed. All other assumptions, bases and methodologies used in the long-term offsite dose calculations remain unchanged. The total dose shown in UFSAR Table 15.6.5-4 does not significantly increase due to the delay in the fission product release. The total amount of radioactivity remains the same and is bounded by the limits established in 10 CFR 100. The dose associated with coolant activity release in the initial 121 seconds of the accident has been determined to be insignificant. Therefore, the proposed change will not result in a significant reduction in the margin of safety.

ENCLOSURE 3 TO
NRC-00-0066

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

REQUEST FOR LICENSE AMENDMENT:

SELECTIVE SCOPE IMPLEMENTATION OF ALTERNATIVE SOURCE TERM INSIGHTS RELATED TO THE ONSET OF GAP ACTIVITY RELEASE TIMING

Attached is a mark-up of the affected UFSAR Sections, indicating the proposed changes (Part 1), and a typed version of the affected UFSAR Sections incorporating the proposed changes (Part 2).

ENCLOSURE 3 TO
NRC-00-0066
PART 1

A MARK-UP OF AFFECTED UFSAR SECTIONS
INDICATING PROPOSED CHANGES

Affected Sections: 15.6.5.5.1, 15.6.5.5.2, 15.6.5.5.3, and 15.6.7

15.6.5.5.1 Fission Product Release From Fuel

It is assumed that 100 percent of the noble gases and 50 percent of the iodine are released from an equilibrium core operating at a power level of 3499 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3, the assumed release of 100 percent of the core noble gas activity and 50 percent of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (refer to Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100 percent of the noble gases and 50 percent of the iodine become airborne. The remaining 50 percent of the iodine is removed by plate out and condensation; therefore, it is not available for airborne release to the environment. ***For primary containment isolation purposes, the activity from the damaged core is assumed to be released into the containment at 121 seconds following the accident. This timing assumption recognizes conclusions derived from the source term studies described in NUREG-1465, Regulatory Guide 1.183 and Reference 4.*** The activity airborne in the containment is presented in Table 15.6.5-2. ***The results in this Table conservatively assume activity released from the core enters the drywell at accident time zero.*** add

15.6.5.5.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the primary containment to the secondary containment by several different mechanisms, as well as discharge to the environment through the standby gas treatment system (SGTS) at an elevated location. The SGTS filter efficiency for iodine removal is assessed as 99 percent. The assumed mechanisms for leakage from the primary containment are discussed below.

a. Containment leakage

   The design basis leak rate of the primary containment and its penetrations to the secondary containment is 0.5 percent per day for the duration of the accident. 96 percent of the activity in the secondary containment escapes to the environment via SGTS which has a 99 percent efficiency. 4 percent of the activity in the secondary containment bypasses SGTS. The duration of exfiltration during the drawdown of the secondary containment is 10 minutes. No credit is taken for mixing and holdup within the secondary containment structure. Figure 15.6.5-1 is an illustration of the release path to the environment.

b. Leakage from engineered safety feature (ESF) components outside the primary containment, which is filtered by the SGTS.

Fission product release to the environment based on the above assumptions is given in Table 15.6.5-3. ***The results in this Table conservatively assume activity released from the core enters the drywell at accident time zero.*** add

Shutdown cooling operation during the 30-day period after a LOCA would involve recirculation of the emergency core coolant water stored in the suppression pool. The emergency core cooling systems used would be the core spray system to cool the reactor core and the RHR system to remove the heat from the emergency coolant. Reactor core cooling with the core spray system is described in Subsection 6.3.2.2.3. Containment cooling with the RHR system is described in Subsection 5.5.7.3.3.

There is no storage of emergency coolant in these systems except in the suppression pool.

The two redundant core spray loops are not connected. The redundant RHR divisions are cross connected for LPCI injection with an isolation valve.

Non seismic piping systems connected to the core spray or RHR systems are seismically qualified up to the first seismic constraint beyond the isolation valve that separates the safety related and non seismic portions of the piping system, as discussed in Section 3.7.3.13. Relief valves on both the RHR and core spray systems discharge back to the suppression pool. The RHR heat exchanger vent lines also drain back to the suppression pool.

The ECCS pump manufacturer's design criteria and technical manuals state that expected leakage for the pump seals is essentially zero. Experience has shown that occasionally seals have a slight leakage when first started; after a short period, this leakage usually ceases.

Edison believes the leakage from the ECCS pump seals to be essentially zero. Industry-wide experience has shown no significant leakage through such pump seals. In spite of this experience and the pump manufacturer's design criteria, which strongly indicate the expected leakage through the seals to be insignificant, the radiological consequences of leakage of water from the emergency cooling water systems have been examined. Hence, in accordance with Appendix B of NRC Standard Review Plan 15.6.5, leakage of ECCS was assumed and a conservative leakage rate of 5-gpm was assigned. It was further assumed that 90 percent of ECCS coolant remains in an unflashed state and that SGTS filter efficiency is 99 percent. The resulting activity in the secondary containment thus undergoes reduction by a factor of a thousand before its release to the environment. The contribution to the total offsite doses as a result of ECCS leakage was 2.67 rem thyroid and 0.0089 rem whole body at the exclusion area, and 1.77 rem thyroid and 0.0021 rem whole body at the low population zone.

15.6.5.5.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6.5-4 and are well within the guidelines of 10 CFR 100. ***Dose associated with coolant activity release in the first 121 seconds of the accident is not included in this Table. Its contribution to the accident dose is insignificant (on the order of 2 rem thyroid at the Exclusion Area Boundary).*** The control room dose analysis is found in Appendix 15A. add

15.6.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown from pipes," ASME PAPER Number 65-WA/HT-1, March 15, 1965

2. F. J. Brutschy, C.R. Hills, N.R. Horton, And A.J. Levin, Behavior of Iodine in Reactor Water During Plant Shutdown and Startup, NEDO-10585.

3. P.P. Stancavage and E. J. Morgan, Conservative Radiological Accident Evaluation - The CONACol Code, NEDO-21143, March 1976.

4. ***General Electric Nuclear Energy, Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR, NEDC-32963A, March 2000.*** add

ENCLOSURE 3 TO
NRC-00-0066
PART 2

AFFECTED UFSAR SECTIONS
INCORPORATING PROPOSED CHANGES

Affected Sections: 15.6.5.5.1, 15.6.5.5.2, 15.6.5.5.3, and 15.6.7

15.6.5.5.1 Fission Product Release From Fuel

It is assumed that 100 percent of the noble gases and 50 percent of the iodine are released from an equilibrium core operating at a power level of 3499 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3, the assumed release of 100 percent of the core noble gas activity and 50 percent of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (refer to Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100 percent of the noble gases and 50 percent of the iodine become airborne. The remaining 50 percent of the iodine is removed by plate out and condensation; therefore, it is not available for airborne release to the environment. For primary containment isolation purposes, the activity from the damaged core is assumed to be released into the containment at 121 seconds following the accident. This timing assumption recognizes conclusions derived from the source term studies described in NUREG-1465, Regulatory Guide 1.183 and Reference 4. The activity airborne in the containment is presented in Table 15.6.5-2. The results in this Table conservatively assume activity released from the core enters the drywell at accident time zero.

15.6.5.5.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the primary containment to the secondary containment by several different mechanisms, as well as discharge to the environment through the standby gas treatment system (SGTS) at an elevated location. The SGTS filter efficiency for iodine removal is assessed as 99 percent. The assumed mechanisms for leakage from the primary containment are discussed below.

a. Containment leakage

   The design basis leak rate of the primary containment and its penetrations to the secondary containment is 0.5 percent per day for the duration of the accident. 96 percent of the activity in the secondary containment escapes to the environment via SGTS which has a 99 percent efficiency. 4 percent of the activity in the secondary containment bypasses SGTS. The duration of exfiltration during the drawdown of the secondary containment is 10 minutes. No credit is taken for mixing and holdup within the secondary containment structure. Figure 15.6.5-1 is an illustration of the release path to the environment.

b. Leakage from engineered safety feature (ESF) components outside the primary containment, which is filtered by the SGTS.

Fission product release to the environment based on the above assumptions is given in Table 15.6.5-3. The results in this Table conservatively assume activity released from the core enters the drywell at accident time zero.

Shutdown cooling operation during the 30-day period after a LOCA would involve recirculation of the emergency core coolant water stored in the suppression pool. The emergency core cooling systems used would be the core spray system to cool the reactor core and the RHR system to remove the heat from the emergency coolant. Reactor core cooling with the core spray system is described in Subsection 6.3.2.2.3. Containment cooling with the RHR system is described in Subsection 5.5.7.3.3.

There is no storage of emergency coolant in these systems except in the suppression pool.

The two redundant core spray loops are not connected. The redundant RHR divisions are cross connected for LPCI injection with an isolation valve.

Non seismic piping systems connected to the core spray or RHR systems are seismically qualified up to the first seismic constraint beyond the isolation valve that separates the safety related and non seismic portions of the piping system, as discussed in Section 3.7.3.13. Relief valves on both the RHR and core spray systems discharge back to the suppression pool. The RHR heat exchanger vent lines also drain back to the suppression pool.

The ECCS pump manufacturer's design criteria and technical manuals state that expected leakage for the pump seals is essentially zero. Experience has shown that occasionally seals have a slight leakage when first started; after a short period, this leakage usually ceases.

Edison believes the leakage from the ECCS pump seals to be essentially zero. Industry-wide experience has shown no significant leakage through such pump seals. In spite of this experience and the pump manufacturer's design criteria, which strongly indicate the expected leakage through the seals to be insignificant, the radiological consequences of leakage of water from the emergency cooling water systems have been examined. Hence, in accordance with Appendix B of NRC Standard Review Plan 15.6.5, leakage of ECCS was assumed and a conservative leakage rate of 5-gpm was assigned. It was further assumed that 90 percent of ECCS coolant remains in an unflashed state and that SGTS filter efficiency is 99 percent. The resulting activity in the secondary containment thus undergoes reduction by a factor of a thousand before its release to the environment. The contribution to the total offsite doses as a result of ECCS leakage was 2.67 rem thyroid and 0.0089 rem whole body at the exclusion area, and 1.77 rem thyroid and 0.0021 rem whole body at the low population zone.

15.6.5.5.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6.5-4 and are well within the guidelines of 10 CFR 100. Dose associated with coolant activity release in the first 121 seconds of the accident is not included in this Table. Its contribution to the accident dose is insignificant (on the order of 2 rem thyroid at the Exclusion Area Boundary). The control room dose analysis is found in Appendix 15A.

15.6.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown from pipes," ASME PAPER Number 65-WA/HT-1, March 15, 1965

2. F. J. Brutschy, C.R. Hills, N.R. Horton, And A.J. Levin, Behavior of Iodine in Reactor Water During Plant Shutdown and Startup, NEDO-10585.

3. P.P. Stancavage and E. J. Morgan, Conservative Radiological Accident Evaluation - The CONACo1 Code, NEDO-21143, March 1976.

4. General Electric Nuclear Energy, Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR, NEDC-32963A, March 2000.