UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 22, 1994

Docket No. 50-341

Mr. Douglas R. Gipson
Senior Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

Dear Mr. Gipson:

SUBJECT: FERMI-2 - ISSUANCE OF AMENDMENT AND EXEMPTION RE: TYPE C TESTING OF PRIMARY CONTAINMENT ISOLATION VALVES (TAC NOS. M86658 AND M86659)

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. NPF-43 for the Fermi-2 facility. The amendment changes the Technical Specifications (TS) in response to your May 24, 1993, letter.

The amendment revises TS 3.4.3.2.d which identifies the allowable leakage for the reactor coolant system pressure isolation valves listed in Table 3.4.3.2-1 by referring to the allowable values which are now listed in the table. The allowable values for the low pressure coolant injection (LPCI) loop A and B isolation valves have changed from 1 gallon per minute (gpm) to 0.4 gpm for each valve. The allowable leakage values for the loop A and B testable check valves have also changed from 1 gpm to 10 gpm for each valve. Additionally, Table 3.6.3-1 has been revised to reflect that the LPCI injection line reverse flow check valves and the 1-inch bypass valves around them are no longer designated as primary containment isolation valves for the purpose of meeting the requirements of General Design Criteria 54 and 55 of Appendix A to 10 CFR Part 50. Table 3.6.3-1 also had a footnote added to reflect an alternative testing method for the LPCI inboard isolation valves. The related bases were also changed to reflect the TS changes.

The Commission has also granted the enclosed exemption regarding the requirements in Appendix J, III.C of 10 CFR Part 50. The exemption allows the use of an alternative testing method for performing Type C local leak rate tests on containment isolation valves in the LPCI lines of the RHR system. We find that granting the exemption is authorized by law, will not present an undue risk to the public health and safety, is consistent with the common defense and security, and meets the special cirumstances described in 10 CFR 50.12(a)(2)(ii).

DF01
111

CP-1

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice. A copy of the exemption is being forwarded to the Office of the Federal Register for publication.

Sincerely,

original signed by

Timothy G. Colburn, Sr. Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 98 to NPF-43
2. Safety Evaluation
3. Exemption

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	OGC	D:PD31	AD:RIII	D:~~DRPW~~
NAME	CJamerson	TColburn:gll	EHoller	LMarsh	JZwolinski	~~JRoe~~ E. ADENSAM
DATE	04/4/94	04/4/94	04/11/94	04/20/94	04/22/94	04/22/94

OFFICIAL RECORD COPY FILENAME: G:\WPDOCS\FERMI\FE86658.AMD

DATED: April 22, 1994

AMENDMENT NO. 98 AND EXEMPTION TO FACILITY OPERATING LICENSE NO. NPF-43-FERMI-2

Docket File
NRC & Local PDRs
PDIII-1 Reading
W. Russell/F. Miraglia
L. Reyes
D. Hagan, MNBB
J. Lieberman, 7H5
J. Roe
J. Zwolinski
L. B. Marsh
C. Jamerson
T. Colburn
OGC-WF
E. Jordan
G. Hill (2), P1-22
C. Grimes
ACRS (10)
OPA
OC/LFDCB
W. Dean, 17G21
M. Phillips, R-III
SEDB

cc: Plant Service list

Mr. Douglas R. Gipson
Detroit Edison Company

Fermi-2

cc:

John Flynn, Esquire
Senior Attorney
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

U.S. Nuclear Regulatory Commission
Resident Inspector Office
6450 W. Dixie Highway
Newport, Michigan 48166

Monroe County Office of Civil
Preparedness
963 South Raisinville
Monroe, Michigan 48161

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Mr. Robert Newkirk
Director - Nuclear Licensing
Detroit Edison Company
Fermi-2
6400 North Dixie Highway
Newport, Michigan 48166

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. NPF-43

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

   A. The application for amendment by the Detroit Edison Company (the licensee) dated May 24, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 98, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance with full implementation within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 98

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

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

REMOVE	INSERT
3/4 4-9*	3/4 4-9*
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 6-28	3/4 6-28
3/4 6-35	3/4 6-35
3/4 6-36*	3/4 6-36*
3/4 6-47	3/4 6-47
3/4 6-48*	3/4 6-48*
B 3/4 4-2	B 3/4 4-2
B 3/4 4-2a	B 3/4 4-2a
B 3/4 6-1	B 3/4 6-1

*Overleaf page provided to maintain document completeness. No changes contained on these pages.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system channel.

b. The primary containment sump flow monitoring system consisting of:

   1. The drywell floor drain sump level, flow and pump-run-time system, and

   2. The drywell equipment drain sump level, flow and pump-run-time system.

c. The drywell floor drain sump level monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE, restore the inoperable detection system to OPERABLE status within 30 days; when the required gaseous radioactive monitoring system is inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours, otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow and drywell floor drain sump level monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. Leakage specified in Table 3.4.3.2-1 at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24 hour period during OPERATIONAL CONDITION 1.

f. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4 hour period during OPERATIONAL CONDITIONS 2 and 3.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valve, or be in at least HOT SHUTDOWN with the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

TABLE 3.4.3.2-1
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	VALVE DESCRIPTION	MAXIMUM LEAKAGE (gpm)
1. RHR System		
E11-F015A	LPCI Loop A Injection Isolation Valve	0.4[(a)]
E11-F015B	LPCI Loop B Injection Isolation Valve	0.4[(a)]
E11-F050A	LPCI Loop A Injection Line Testable Check Valve	10
E11-F050B	LPCI Loop B Injection Line Testable Check Valve	10
E11-F008	Shutdown Cooling RPV Suction Outboard Isolation Valve	1
E11-F009	Shutdown Cooling RPV Suction Inboard Isolation Valve	1
E11-F608	Shutdown Cooling Suction Isolation Valve	1
2. Core Spray System		
E21-F005A	Loop A Inboard Isolation Valve	1
E21-F005B	Loop B Inboard Isolation Valve	1
E21-F006A	Loop A Containment Check Valve	1
E21-F006B	Loop B Containment Check Valve	1
3. High Pressure Coolant Injection System		
E41-F007	Pump Discharge Outboard Isolation Valve	1
E41-F006	Pump Discharge Inboard Isolation Valve	1
4. Reactor Core Isolation Cooling System		
E51-F012	Pump Discharge Isolation Valve	1
E51-F013	Pump Discharge to Feedwater Header Isolation Valve	1

(a) External Leakage from this valve shall be limited to 5 ml/min.

TABLE 3.4.3.2-2
REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F050A & B	RHR LPCI	≤ 449
E11-F008, F009, F608	RHR Shutdown Cooling	≤ 135
E21-F005A & B, E21-F006A & B	Core Spray	≤ 452
E41-F006, F007	HPCI	≤ 71
E51-F012, F013	RCIC	≤ 71

TABLE 3.6.3-1 (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves(e)	
1. Main Steam Isolation Valves (MSIV) Leakage Control Valves	
B21-F434	NA
2. RHR Shutdown Cooling Suction Inboard Isolation Valve Bypass Valve(q)	
E11-F608	NA
3. LPCI Inboard Isolation Valves(f)(s)	
Loop A: E11-F015A Loop B: E11-F015B	NA
4. RHR Pumps Recirculation Motor Operated Valves(b)(g)	
Pumps A/C: E11-F007A Pumps B/D: E11-F007B	NA
5. Warmup and Flush Line Isolation Valve(b)	
E11-F026B	NA
6. Reactor Protection System Instrumentation Isolation Valves	
Division I: E11-F412 E11-F413 Division II: E11-F414 E11-F415	NA
7. RHR Pump Torus Suction Isolation Valves(b)	
Pump A: E11-F004A Pump B: E11-F004B Pump C: E11-F004C Pump D: E11-F004D	NA

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves	
1. Main Feedwater Reverse Flow Check Valves	
B21-F010A	NA
B21-F010B	
B21-F076A	
B21-F076B	
2. Deleted	
3. RHR Heat Exchanger Relief Valves(b)	
E11-F001A	NA
E11-F001B	
4. RHR Heat Exchanger Outlet Line Relief Valves(b)(p)	
E11-F025A	NA
E11-F025B	
5. RHR Pump Suction From Recirc Piping Reverse Flow Check Valve	
E11-F408	NA
6. RHR Shutdown Cooling Suction Relief Valve(b)(p)	
E11-F029	NA
7. RHR Pump Torus Suction Relief Valves(b)(p)	
E11-F030A	NA
E11-F030B	
E11-F030C	
E11-F030D	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
8. Core Spray Loop Containment Reverse Flow Check Valves	NA
E21-F006A	
E21-F006B	
9. Core Spray Loop Pump Suction Relief Valves(b)(p)	NA
E21-F032A	
E21-F032B	
10. Core Spray Loop Pump Discharge Pressure Relief Valves(b)	NA
E21-F011A	
E21-F012A	
E21-F011B	
E21-F012B	
11. Excess Flow Check Valves(r)	NA
a. Jet Pump Instrumentation	
B21-F513A	
B21-F513B	
B21-F513C	
B21-F513D	
B21-F514A	
B21-F514B	
B21-F514C	
B21-F514D	
B21-F515A	
B21-F515B	
B21-F515C	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS (Continued)

(k) Will automatically close when a) RCIC Turbine Steam Stop Valve E51-F045 closes or b) RCIC Turbine Governor Trip and Throttle Valve E51-F059 closes.

(l) Will automatically close as a result of the conditions listed in Note (k) above, as well as when RCIC flow is greater than 130 gpm.

(m) These valves are actuated by remote manual key-locked switches and will cut the TIP cable and seal off the TIP guide tube when actuated. These valves are squib-fired.

(n) May be closed remotely as a secondary actuation mode to reverse flow.

(o) Valves realign automatically on a reactor scram signal.

(p) Thermal relief valves.

(q) Locked closed.

(r) Not subject to Type C leakage tests.

(s) Hydrostatically tested in accordance with Specification 4.4.3.2.2 in lieu of the requirements of Specification 4.6.1.2.

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be closed and OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the above required vacuum breakers inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more suppression chamber - drywell vacuum breakers open, close the open vacuum breaker(s) within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one of the position indicators of any suppression chamber - drywell vacuum breakers inoperable, verify that all other vacuum breakers are closed within 2 hours and:

   1. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by demonstrating the other indicator to be OPERABLE within 2 hours and at least once per 14 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the drywell-to-suppression chamber ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 14 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one of the closed position indicators of one or more suppression chamber - drywell vacuum breaker(s) indicating open and the redundant closed position indicator indicating closed after a suppression chamber - drywell vacuum breaker opening as a result of a steam release, within 24 hours, cycle the applicable valve(s) to determine which of the redundant indicators is OPERABLE.

3/4.4.4 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that contain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits. The additional limit placed upon the rate of increase in UNIDENTIFIED LEAKAGE in OPERATIONAL CONDITION 1 meets the NRC Staff guidance in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." The applicability of the Generic Letter 88-01 limit to OPERATIONAL CONDITION 1 only ensures that the expected increases in UNIDENTIFIED LEAKAGE experienced during reactor vessel heatup and pressurization during startup do not cause unwarranted entries into the applicable ACTION statement. The rate of increase in UNIDENTIFIED LEAKAGE limit in OPERATIONAL CONDITIONS 2 and 3 ensures that the above service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping is monitored during reactor startup prior to reactor vessel heatup and pressurization. The surveillance interval for determination of UNIDENTIFIED LEAKAGE in OPERATIONAL CONDITION 1 meets the guidance in Supplement 1 to Generic Letter 88-01.

The purpose of the RCS interface valves leakage pressure monitors (LPMs) is to provide assurance of the integrity of the Reactor Coolant System pressure isolation valves which form a high/low pressure boundary. The LPM is designed to alarm on increasing pressure on the low pressure side of the high/low pressure interface to provide indication to the operator of abnormal interface valve leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.3.2 OPERATIONAL LEAKAGE (Continued)

A reduced leakage acceptance criteria and an external leakage acceptance criteria are specified for the LPCI Injection Isolation Valve, E11-F015 A and B, to assure adequate water is maintained inboard of these valves such that the associated primary containment penetration can be classified as a water tested penetration under Appendix J to 10CFR50.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6 CONTAINMENT SYSTEMS
BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing, testing the airlocks after each opening, testing the Low Pressure Coolant Injection Inboard Isolation Valves, and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated May 24, 1993, the Detroit Edison Company (DECo or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment revises TS 3.4.3.2.d which identifies the allowable leakage for the reactor coolant system (RCS) pressure isolation valves listed in Table 3.4.3.2-1 by referring to the allowable values which are now listed in the table. The allowable values for the low pressure coolant injection (LPCI) loop A and B isolation valves have changed from 1 gallon per minute (gpm) to 0.4 gpm for each valve. The allowable leakage values for the loop A and B testable check valves have also changed from 1 gpm to 10 gpm for each valve.

Additionally, Table 3.6.3-1 has been revised to reflect that the LPCI injection line reverse flow check valves and the 1-inch bypass valves around them are no longer designated as primary containment isolation valves (CIVs) for the purpose of meeting the requirements of General Design Criteria (GDC) 54 and 55 of Appendix A to 10 CFR Part 50. Table 3.6.3-1 also had a footnote added to reflect an alternative testing method for the LPCI loop A and B isolation valves. The licensee also requested an exemption from the related Type C testing requirements of Appendix J to 10 CFR Part 50 for these isolation valves.

2.0 DISCUSSION

On each of the two LPCI injection lines, three valves are currently specified as CIVs in TS Table 3.6.3-1, Primary Containment Isolation Valves. A motor-operated gate valve, E11-F015, is installed outboard of containment. E11-F015 is remotely operated from the control room and is installed in the 24-inch injection line which connects to the RCS in the reactor recirculation loop.

Also installed on the 24-inch injection line inside containment is a check valve, E11-F050. E11-F050 is installed to prevent reverse flow out of containment through the injection line. A 1-inch bypass line is installed around E11-F050 for the purpose of system warmup for the residual heat removal (RHR)-shutdown cooling mode. A solenoid-operated globe valve, E11-F610, is normally closed except during system warmup and is remotely operated from the control room.

The two piping penetrations are designated as primary containment penetrations X-13A and X-13B. The valves associated with each penetration have a suffix of "A" or "B" added to their labels; thus, valve E11-F015A is in the line going through penetration X-13A, and so on. The suffixes are not used in most of this discussion.

These CIVs are subject to Type C local leak rate testing in accordance with Appendix J. The Type C tests are performed with a gas at 56.5 psig, which is Pa, the calculated peak containment internal pressure during a design basis accident.

E11-F015 and E11-F050 are also designated in TS Table 3.4.3.2-1, "Reactor Coolant System Pressure Isolation Valves," as pressure isolation valves. As such, these valves form a boundary between the high pressure RCS piping and low pressure RHR piping.

GDC 55, in Appendix A to 10 CFR Part 50, requires these penetrations to each have two CIVs, either automatic or locked closed, one inside and one outside containment, unless alternative provisions are found acceptable on some other defined basis. In this case, the outside valve, E11-F015, is a remote manual valve rather than automatic or locked closed, because LPCI is required to operate for core cooling during an accident. The inside containment configuration meets the GDC explicitly since E11-F050 is automatically closed by reverse flow out of containment and E11-F610 is locked closed. No automatic containment isolation signal is used to isolate these valves due to the essential nature of the RHR-LPCI mode.

However, E11-F050 has repeatedly failed the Type C test requiring repair and retesting. Although the licensee has been able to repair and successfully retest the valve, the resultant radiological and safety concerns caused the licensee to review the use of E11-F050 as a CIV under these circumstances.

E11-F050 is located in a hazardous area in the drywell. The area is also within a significant radiation field. For example, during the last refueling outage, 10.9 person-rem were expended in maintenance and testing on E11-F050A and B. In addition, this type of work also extends the outage duration and increases the unavailability of one of the decay heat removal systems during the outage.

As a result of the review of the design basis for these penetrations, the licensee has determined that there exists an alternate basis for meeting the containment isolation provisions of GDC 55 without the attendant valve testing problems associated with the current means of compliance with GDC 55. This alternate basis examines aspects of the RHR system which had not been included in previous evaluations. It results in E11-F050 and E11-F610 no longer being CIVs. Further, the licensee has developed a basis for the Appendix J exemption which justifies alternate leak rate testing provisions. The new testing requirements will result in a reduction in occupational radiation exposure and improvements in industrial safety while continuing to assure containment integrity. None of the valves would any longer receive Type C tests with air as the test medium, although E11-F015 would have its external leakage measured (with water as the test medium) and limited to a certain value.

3.0 EVALUATION

Two aspects of the RHR system form the basis for the proposed TS changes and exemption; (1) it is a closed system outside containment, and (2) the penetrations will be water sealed during a loss-of-coolant accident (LOCA). These aspects are discussed below.

3.1 Closed System Outside Containment

The staff's Standard Review Plan (SRP), NUREG-0800, in Section 6.2.4, "Containment Isolation System," provides several "other defined bases" for differing from the explicit requirements of GDC 55. Subsection II.6.e allows only a single CIV, outside containment, if the system is closed outside containment and certain other provisions are met.

The RHR system is a closed system outside containment. It can accommodate a single active failure and still maintain containment integrity. It is also protected against the effects of missiles and pipe whip. The system is designed to seismic Category I standards, is classified as Quality Group B or better, and is designed to meet or exceed the maximum temperature and pressure of the containment.

The system is also included in the American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection Program and receives the required non-destructive examinations for Class 2 piping. These programs require the system to be inspected at pressure and any visible leakage to be promptly repaired.

The RHR system is also subjected to the inspections required by the Fermi 2 system leakage reduction program. This program is a commitment made by the licensee in response to NUREG-0737, "Clarification of TMI Action Plan Requirements," to conduct inspections to reduce and maintain leakage to as low as practical levels from systems outside of the primary containment that could or would contain highly radioactive fluids during or after a severe transient or accident. The inspections are done on an 18-month frequency and the acceptance criterion for the RHR system is 40 ml/min external leakage for each RHR division.

The piping from the inboard check valves E11-F050A and B to valves E11-F015A and B conforms with ASME Section III, Class 1 requirements. Since this piping fulfills the design requirements stipulated in Branch Technical Position MEB 3-1, no pipe breaks or cracks are postulated.

The RHR system is currently considered an acceptable closed system outside containment for the purposes of meeting GDC 56 for penetrations X-223A through D. These penetrations are for the RHR pump suction from the suppression pool.

The above design provisions are in accordance with SRP 6.2.4, II.6.e. Therefore, penetrations X-13A and B need only have one CIV each, outside containment, to comply with GDC 55, because this is acceptable on "some other defined basis," said basis being defined by the SRP.

The licensee proposes to no longer designate E11-F050A and B and E11-F610A and B as CIVs. However, these valves will remain in place and will be tested as high/low pressure isolation valves in accordance with TS 4.4.3.2.2. The staff finds this to be acceptable, as discussed above. This will, of course, eliminate the requirement to Type C test the valves, because Type C testing applies only to CIVs.

3.2 Water Seal

CIV leakage can be categorized as either through the valve seat or external to the valve (such as a stem or bonnet leak). The licensee has provided analyses which show that water seals would exist at E11-F015, the CIV outside containment, which would prevent the leakage of containment atmosphere through the valve, via either mode, during a LOCA, despite the most limiting single active failure. The two types of water seals are discussed below.

3.2.1 Through-seat water seal

Operation of the RHR pumps assures that any through-seat leakage for valve E11-F015 will be water leakage inward towards the containment despite any single active failure. This water seal would prevent the leakage of containment atmosphere through the seat and out of containment. The following design features assure the water seal:

1. Each division of RHR has two pumps each of which are fed from a separate diesel generator.

2. The two divisions for RHR are cross-connected by a single header through the E11-F010 valve which is normally kept open.

3. Valves E11-F015A and B, E11-F010, and the two recirculation pump valves are electrically fed from the swing bus. The swing bus design assures power availability to each of the above valves in case one of the divisional power supplies is lost.

4. One or more of the RHR pumps will be run following an accident for either LPCI injection or suppression pool cooling for at least a period of 30 days. In some cases, both functions will be carried out simultaneously. The LPCI cross-tie valve will be closed in these cases; however, then RHR pumps will be operating in both divisions.

5. In the long term, the RHR pumps draw suction from the suppression pool, providing a continuous or unlimited supply of water for at least 30 days.

A detailed analysis is provided in the licensee's submittal, but it can be seen that a water seal, pressurized to greater than 1.1 Pa (62.15 psig), would be provided at the E11-F015 valve for at least 30 days following the onset of a LOCA, despite the most limiting single active failure, preventing through-seat leakage of containment atmosphere. However, one or both of the valves E11-F015A and B may be closed during a LOCA, and this water seal might not prevent leakage external to the valve (e.g., stem or bonnet leakage); thus, the need for the following analysis.

3.2.2 External leakage water seal

The configuration of the LPCI injection lines is such that a water leg would be present inboard of E11-F015 post-accident. The licensee has provided in its submittal a conservative and very detailed analysis to show that the water leg would be at least 23.4 ft. long, in a 24-inch diameter pipe. The staff has reviewed this analysis and finds it to be acceptable.

The licensee proposes a limit of 5 ml/min external leakage from E11-F015. It can be seen that, at that leak rate, the water leg would last more than 30 days, so that no containment atmosphere would leak out of containment during that period. Through-seat leakage would not deplete the water leg, because, as discussed above, through-seat leakage would be water in toward containment, not out.

External leakage testing would be performed as part of the pressure isolation valve (PIV) leak rate testing required by TS at 18 month intervals, with water as the test medium and at 1045 psig.

This pressure is much greater than any accident pressure. However, this is conservative since higher pressures tend to yield a greater external leakage. Water is the appropriate test medium because the valve would be water sealed. Valve through-seat leakage is also measured by this test and limited to 0.4 gpm by the TS, but this limit is not necessary for the maintenance of either water seal.

Although the external leakage water seal would prevent atmosphere from leaking out of containment, it does not satisfy the requirements for a water seal contained in Appendix J. Appendix J allows water-sealed valves to be excepted from normal Type C testing with air, but it requires the water seal to be pressurized to at least 1.1 Pa during an accident. The water leg inboard of E11-F015 does not meet this requirement, nor does it meet the spirit of the rule, wherein a water seal would at worst make water go into containment, rather than let it leak out. Nevertheless, the staff finds that the licensee's analyses of the water seals provide sufficient assurance that containment atmosphere leakage out of containment will be prevented during an accident, to justify the granting of the requested exemption from Type C testing of valves E11-F015A and B with air as the test medium. In lieu of that, the licensee will measure external valve leakage in conjunction with PIV leak rate testing and limit it to 5 ml/min.

3.3 Pressure Isolation Valve Allowable Leakage Changes

The subject valves are a pair of valves in each of two redundant LPCI lines which also function during RHR. The inboard valve of each pair is a 24" Anchor Darling swing check valve (E11-050 A/B) inside containment. The outboard valve of each pair is a 24" motor-operated flexible wedge gate valve (E11-F015 A/B) outside containment. The TS proposes to change the allowable leakage valves for the pressure isolation function from the current maximum of 1 gpm allowable leakage, to 10 gpm and 0.4 gpm, for the check valves and gate valves, respectively.

The current Fermi-2 TS require PIV leakage to be limited to 1 gpm. This limit is consistent with the limits imposed by the NRC prior to 1985 at which time the limits were changed in Standard Technical Specifications to 0.5 gpm per nominal inch of valve size, up to 5 gpm. The original leak rate acceptance criterion was promulgated by the NRC in response to the requirement to establish allowable leakage rates for the "Event V" described in WASH-1400. The "Event V" scenario was the failure of two check valves in series subjecting a low pressure system outside of a pressurized-water reactor (PWR) containment to full reactor pressure, rupturing the low pressure piping and causing a LOCA which bypassed containment. The leakage criterion for "Event V" valves was 1.0 gpm, with leakage between 1.0 and 5.0 gpm being acceptable if the leakage did not increase by an amount that reduced the margin between the measured leakage rate and the maximum permissible (5 gpm) by 50% or greater.

For subsequent near-term operating license (NTOL) applications, the requirement for inservice leak testing was extended to all PIVs (first and second valve in series leading away from reactor coolant pressure) either inside or outside containment and including both boiling-water reactor and PWR plants. In general, for operating reactors, the leak rate acceptance criterion for "Event V" valves was imposed on all PIVs. For NTOL reviews, a maximum leak rate of 1 gpm, generally without qualification, was imposed for all PIVs. The stricter acceptance criterion was imposed on the newer plants because it was considered that the valves could meet the standard, being newer and in better condition. The limit was based on a conservative assessment of the capacity of the pressure relief systems in most of the plants.

The staff later determined that the 1 gpm acceptance criterion was not an indicator of imminent accelerated deterioration of valves or of potential valve failure. Regardless of the leak rate allowed for each PIV, plant TS limit the allowable leakage from the RCS as a whole. The system restriction effectively eliminates the possibility of large increases in leakage from a number of valves in the RCS. Increasing the allowable PIV leakage would, therefore, have no impact on the total allowable RCS leakage.

The NRC contracted a study of PIV leak test requirements (EGG-NTAP-6175, "Inservice Leak Testing of Primary Pressure Isolation Valves," February 1983) which presented an assessment of PIV inservice leak testing and allowable leakage limits. The report recommended that the owner be allowed the option of a higher allowance of leakage for specific valves if justified by an analysis of overpressure protection and radiological processing capability showing that the ASME *Boiler and Pressure Vessel Code* (the Code), Section III, and the plant safety analysis conclusions are not violated by the higher leakage allowance. The primary objective of the staff in allowing higher leak rates was to decrease the time spent on unnecessary maintenance on the valves which attributes to faster deterioration of the valves and increased exposure to personnel. The choice of an absolute maximum allowable leakage rate of 5 gpm was based on experience gained from the plant using 5 gpm to comply with "Event V" orders. Allowable leak rates above 5 gpm were not considered conservative because of a lack of experience.

The EG&G report identified the following reasons for limiting the leakage of reactor coolant into lower pressure interfacing systems, such as the LPCI system:

(1) Leakage from the RCS, together with other sources may exceed the flow capacity of the pressure relief system, causing overpressurization of a lower pressure system.

(2) A large allowance of identified leakage through primary pressure isolation valves may make it more difficult for leakage detection systems to identify small but important increases in unidentified leakage.

(3) Leakage of radiologically contaminated RCS water may exceed the processing capacity of waste water cleanup systems.

(4) This allowed breach of containment may increase the probability of uncontrolled fission product release under certain accident conditions.

EGG-NTAP-6175 concluded that the TS limiting values for leakage of 1 to 5 gpm, or 0.5 gpm per inch of diameter up to 5 gpm, are acceptable. An allowance of more than 5 gpm leakage was not recommended because (1) the leakage may not be accounted for in the ASME *Boiler and Pressure Vessel Code*, Section III, overpressure protection analysis, (2) the larger leakage allowances would tend to mask the detection of unidentified leakage from the primary system, which for Fermi-2 is limited to 5 gpm per TS, and (3) the leakage should be limited to limit fission product distribution from the primary coolant to plant systems outside the containment. As proposed, however, the total leakage from the two in-series valves is maintained below the previous total allowed (0.4 gpm versus 1 gpm), which results in a more conservative limit for all cases except inadvertent opening of the motor-operated gate valve (which is interlocked closed below shutdown primary system pressure). Even assuming the maximum leakage of 10 gpm, each of these three factors are considered and discussed below in relation to the proposed TS change:

(1) The analysis prepared by DECo addresses the overpressure protection analysis of the RHR system, indicating that a 1-inch relief valve on each of the low pressure side of the pressure isolation valves has a capacity of 290 gpm at 450 psig. The Fermi-2 updated final safety analysis report (UFSAR), Section 5.5.7.3.5, states that the "reactor coolant system pressure boundary isolation valve leakage is accommodated by 1-in. relief valves. This size of the valve is considered large enough to accommodate any postulated leakage." The analysis indicates that the UFSAR evaluation

is based on a PIV leakage test acceptance criteria of 10 gpm. Therefore, a leakage limit of 10 gpm appears to be acceptable from an overpressure protection concern.

(2) The analysis submitted by DECo did not specifically address the possibility of masking the detection of unidentified leakage from the primary system due to an increase of the allowable leakage limit of the check valves; however, by specifying a lower limit for the gate valve (0.4 gpm), which is interlocked closed with RCS pressure above approximately 450 psig, the pair of valves will maintain leakage below a level of concern for increasing the possibility of masking leakage. The total unidentified RCS leakage allowed by TS is a conservative 5 gpm. Therefore, an increase in the leakage limit for the check valve does not create a safety concern.

(3) For concerns of limiting leakage outside containment due to PIV leakage, the DECo analysis for the TS change discusses this aspect in terms of containment isolation issues; however, the results are applicable to PIV leakage as well. The two series isolation valves provide redundancy for each of the two RHR injection lines. The motor-operated gate valves are interlocked with a pressure switch that prohibits opening of the valve if the recirculation pressure exceeds the shutdown range (Fermi 2 UFSAR Section 5.5.7.3.5). As noted in item (1) above, the relief RHR valve is of a size to relieve the maximum allowable leakage of either valve (10 gpm) with the discharge to inside containment. The RHR system forms a closed system outside containment, with a water seal maintained as a result of the physical configuration of the LPCI loop selection feature of the RHR system. The RHR system is protected against the effects of missiles and pipe whip, is subject to inservice inspection for ASME Class 2 piping, and is included in the leakage reduction program established in accordance with NUREG-0737 to reduce leakage of primary water outside containment. The piping between the inboard check valve and the outboard motor-operated gate valve conforms to ASME Class 1 piping requirements. The combination of these design features and monitoring programs serve to limit leakage outside containment; therefore, the increase in allowable leakage will not create a safety concern.

Based on the foregoing evaluation, the staff finds that the licensee has provided adequate bases for changing the allowable leakage values for RCS pressure isolation valves listed in Table 3.4.3.2-1, specifically the LPCI loop A and B isolation valves and the loop A and B testable check valves, from 1 gpm to 0.4 gpm and 10 gpm, respectively. Additionally, the staff finds that the licensee has provided adequate alternative bases for meeting GDC 54 and 55 and that the LPCI reverse flow check valves and 1-inch bypass valves may be removed from the list of primary containment isolation valves in 3.6.3-1.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 46227). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.32, an environmental assessment of the exemption from certain requirements of 10 CFR Part 50, Appendix J, related to these actions was published in the Federal Register on April 21, 1994 (59 FR 19028). Accordingly, the Commission has determined that the issuance of this exemption will not result in any environmental impacts beyond those evaluated in Fermi-2's Final Environmental Statement.

6.0 CONCLUSION

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

Principal Contributors: J. Pulsipher
P. Campbell

Date: April 22, 1994

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
DETROIT EDISON COMPANY	)	Docket No. 50-341
(Fermi 2)	)	

EXEMPTION

I.

Detroit Edison Company (the licensee) is the holder of Facility Operating License No. NPF-43 which authorizes operation of the Fermi 2 Nuclear Plant at steady-state reactor power levels not in excess of 3430 megawatts thermal. The Fermi 2 facility is a boiling water reactor located at the licensee's site in Monroe County, Michigan. The license provides, among other things, that it is subject to all rules, regulations, and Orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

II.

Paragraph III.C of Appendix J to 10 CFR Part 50 requires, in part, that valves, unless pressurized with fluid (e.g., water, nitrogen) from a seal system, shall be tested by pressurizing with air or nitrogen at a test pressure of Pa (56.5 psig), the calculated peak containment internal pressure as a result of the design basis accident. Further, the combined leakage rate of all penetrations and valves subject to Type B and C testing shall be less than 0.60 La (La is the maximum allowable leakage rate at Pa). Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded, provided the leakage rates for these valves do not exceed the Technical Specification leakage requirements and the seal system fluid inventory is sufficient to ensure the sealing function for 30 days following an accident at a pressure of 1.10 Pa.

Pursuant to 10 CFR 50.12(a), the NRC may grant exemptions from the requirements of the regulations (1) which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) where special circumstances are present.

III.

By letter dated May 24, 1993, the licensee requested an exemption from the requirements of 10 CFR Part 50, Appendix J, III.C for performing Type C integrated leak rate tests of the containment isolation valves in the low pressure coolant injection (LPCI) lines of the residual heat removal (RHR) system. The valves in question are the loop A and B LPCI isolation valves, which are motor-operated gate valves, outboard of containment and which are operable from the control room and are designated as E11-F015 A and B. The licensee proposed, as alternative testing, an external leakage test, with water as the test medium, at a pressure of 1045 psig with an allowable leakage value of 5 milliliters per minute (ml/min). The licensee also provided justification in its May 24, 1993, letter to reclassify the inboard containment LPCI valve configuration (which consists of a reverse flow swing check valve with a 1-inch, locked closed, solenoid-operated bypass valve) as other than containment isolation valves and thus no longer subject to Type C testing.

The staff evaluated the licensee's proposal for reclassification of the inboard containment valves and concluded that the licensee's proposal met the guidance in the Standard Review Plan (SRP), NUREG-0800, Section 6.2.4 for differing from the explicit requirements of General Design Criterion 55 in 10 CFR Part 50, Appendix A for containment isolation valves (CIVs). Subsection II.6.e allows only a single CIV outside containment, if the system is closed outside containment and certain other criteria are met. Details concerning the staff's review are contained in the staff's safety evaluation dated April 22, 1994.

Two aspects of the RHR system form the basis for the proposed exemption. It is a closed system outside of containment, and the containment penetrations will be water sealed during a loss of coolant accident (LOCA). The licensee's analyses showed that a water seal, pressurized to greater than 1.1 Pa (62.15 psig), would exist outboard of the LPCI CIV for at least 30 days following the design basis LOCA despite the most limiting single active failure. However, if one or both of the LPCI CIVs is shut, that water seal might not prevent external valve leakage (valve stem or bonnet leakage). The licensee also showed that a water seal would exist inboard of the LPCI CIVs following a design basis LOCA. The licensee's analyses demonstrated that the volume of the water seal is sufficient to last for greater than 30 days assuming the leakage limit proposed in their alternative testing acceptance criteria. The licensee's analyses also showed that through seat leakage of the LPCI CIVs would be in toward containment and would not deplete the water seal.

Although the external leakage water seal would prevent atmosphere from leaking out of containment, it does not satisfy the requirements for a water seal contained in Appendix J. Appendix J allows water sealed valves to be excepted from the normal Type C testing with air, but it requires that the water seal be pressurized to at least 1.1 Pa during an accident. The water leg inboard of the LPCI CIVs does not meet this requirement. Nevertheless, the staff has determined that the licensee's analyses of the water seals provide sufficient assurance that containment atmosphere leakage out of containment will be prevented during an accident to justify granting the requested exemption from Type C testing of the LPCI CIVs with air as the test medium. The licensee's alternative test will measure external valve leakage with a limit of 5 ml/min using water as the test medium at a pressure of 1045 psig.

IV.

Accordingly, the Commission concluded that the licensee's proposed alternative testing plan and analyses provide sufficient assurance that the containment atmosphere would not leak out of containment through the LPCI CIVs during a design basis accident and that containment integrity will be maintained by granting the proposed exemption.

The special circumstances for granting this exemption pursuant to 10 CFR 50.12 have also been identified. As stated in part in 10 CFR 50.12(a)(2)(ii), special circumstances are present when application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The purpose of Section III.C of Appendix J is to measure containment isolation valve leakage rates. This leakage, when summed with the allowable Type A and Type B leakage is limited to a value which ensures overall containment integrity in preventing the uncontrolled release of radioactivity to the environment. The licensee has demonstrated through analyses and by proposing alternative testing criteria, that containment integrity will be maintained. Consequently, the Commission concludes that the special circumstances of 10 CFR 50.12(a)(2)(ii) exist in that application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule.

V.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, this exemption as described in Section III above is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The Commission further determines that special circumstances as provided in 10 CFR 50.12(a)(2)(ii) are present justifying the exemption.

Therefore, the Commission hereby grants an exemption as described in Section III above from the requirements in 10 CFR Part 50, Appendix J, III.C. for performing Type C containment integrated leak rate tests of the CIVs in the LPCI lines of the RHR system and approves the licensee's alternative testing plan.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will have no significant impact on the environment (59 FR 19028).

This exemption is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Acting Director
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 22nd day of April 1994