

Template NRR-058

March 14, 2000

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

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: REVISED EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS (TAC NO. MA7373)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated December 17, 1999, as supplemented January 26, 2000.

The amendment revises TS Surveillance Requirement 3.6.1.3.9 to allow a representative sample of reactor instrumentation line excess flow check valves to be tested every 18 months, instead of testing each excess flow check valve every 18 months. The associated inservice testing request for relief is being reviewed separately under TAC No. MA7374.

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

Sincerely,

/RA/

Andrew J. Kugler, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No.137 to NPF-43
2. Safety Evaluation

OFFICE FILE CENTER COPY

cc w/encls: See next page

DISTRIBUTION

File Center ACRS WBeckner MReinhart
PUBLIC OGC GHill(2) MRubin
PDIII-1 Reading AVeigel, RIII

TO RECEIVE A COPY OF THIS DOCUMENT, INDICATE "C" IN THE BOX

*SEE PREVIOUS CONCURRENCE

OFFICE	PDIII-1/PM	C	PDIII-1/LA	C	SPLB/SC*		SPSB/C*		OGC*		PDIII-1/SC	C
NAME	AKugler		RBouling		GHubbard		RBarrett		SHom		CCraig	
DATE	03/13/00		03/13/00		02/25/00		02/24/00		03/02/00		03/13/00	

DOCUMENT NAME: G:\PDIII-1\FERMI\AMD-A7373.wpd

OFFICIAL RECORD COPY

DF01

March 14, 2000

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

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: REVISED EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS (TAC NO. MA7373)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated December 17, 1999, as supplemented January 26, 2000.

The amendment revises TS Surveillance Requirement 3.6.1.3.9 to allow a representative sample of reactor instrumentation line excess flow check valves to be tested every 18 months, instead of testing each excess flow check valve every 18 months. The associated inservice testing request for relief is being reviewed separately under TAC No. MA7374.

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

Sincerely,

/RA/

Andrew J. Kugler, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No.137 to NPF-43
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

File Center
PUBLIC
PDIII-1 Reading

ACRS
OGC
AVegel, RIII

WBeckner
GHill(2)

MReinhart
MRubin

TO RECEIVE A COPY OF THIS DOCUMENT, INDICATE "C" IN THE BOX

*SEE PREVIOUS CONCURRENCE

OFFICE	PDIII-1/PM	C	PDIII-1/LA	C	SPLB/SC*	SPSB/C*	OGC*	PDIII-1/SC
NAME	AKugler <i>AK</i>		RBouling <i>RB</i>		GHubbard	RBarrett	SHom	CCraig <i>CC</i>
DATE	03/13/00		03/13/00		02/25/00	02/24/00	03/02/00	03/13/00

DOCUMENT NAME: G:\PDIII-1\FERMI\AMD-A7373.wpd

OFFICIAL RECORD COPY



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

March 14, 2000

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

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: REVISED EXCESS FLOW CHECK
VALVE SURVEILLANCE REQUIREMENTS (TAC NO. MA7373)

Dear Mr. Gipson:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated December 17, 1999, as supplemented January 26, 2000.

The amendment revises TS Surveillance Requirement 3.6.1.3.9 to allow a representative sample of reactor instrumentation line excess flow check valves to be tested every 18 months, instead of testing each excess flow check valve every 18 months. The associated inservice testing request for relief is being reviewed separately under TAC No. MA7374.

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

Sincerely,

Andrew J. Kugler, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No. 137 to NPF-43
2. Safety Evaluation

cc w/encls: See next page

Fermi 2

cc:

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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P. O. Box 30630 CPH Mailroom
Lansing, MI 48909-8130

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

Monroe County Emergency Management
Division
963 South Raisinville
Monroe, MI 48161

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

Norman K. Peterson
Director, Nuclear Licensing
Detroit Edison Company
Fermi 2 - 280 TAC
6400 North Dixie Highway
Newport, MI 48166

November 1999



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. 137
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 December 17, 1999, as supplemented January 26, 2000, 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.

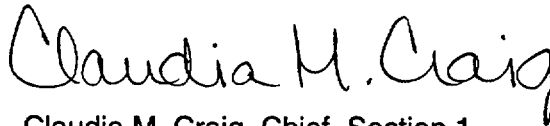
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 137 , 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 14, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE

3.6-16
B 3.6.1.3-15
B 3.6.1.3-16
B 3.6.1.3-17
B 3.6.1.3-18

INSERT

3.6-16
B 3.6.1.3-15*
B 3.6.1.3-16*
B 3.6.1.3-17*
B 3.6.1.3-18*

* Bases pages are controlled by the licensee under Technical Specification 5.5.10, "Technical Specifications (TS) Bases Control Program." These pages are included with the amendment for information only.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days <u>AND</u> Once within 92 days after opening the valve
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.9	Verify a representative sample of reactor instrumentation line EFCVs actuates on a simulated instrument line break to restrict flow.	18 months
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)SR 3.6.1.3.7

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

SR 3.6.1.3.8

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

SR 3.6.1.3.9

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.6.1.3.10

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

SR 3.6.1.3.11

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

Secondary containment bypass leakage is also considered part of L_s .

SR 3.6.1.3.12

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

SR 3.6.1.3.13

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1 gpm times the number of valves per penetration, not to exceed 3 gpm, when tested at $1.1 P_s$ (≥ 62.2 psig). Additionally, a combined leakage rate limit of ≤ 5 gpm when tested at $1.1 P_s$ (≥ 62.2 psig) is applied for all hydrostatically tested PCIVs that penetrate containment. The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by Primary Containment Leakage Rate Testing Program.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Table 6.2-2
3. 10 CFR 50, Appendix J, Option B.
4. UFSAR, Section 6.2.
5. UFSAR, Section 15.6.2.
6. GE BWROG B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 137 FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By application dated December 17, 1999, as supplemented January 26, 2000, the Detroit Edison Company (DECo or the licensee) requested changes to the Technical Specifications (TSs) for Fermi 2. The proposed changes would relax the surveillance frequency for excess flow check valves (EFCVs) in reactor instrumentation lines to allow testing of a "representative sample" of EFCVs every 18 months, rather than testing each EFCV every 18 months, as currently required by the TS. The licensee proposes to test approximately 20 percent of the EFCVs each 18 months such that each EFCV will be tested at least once every 10 years (nominal). The licensee states that its basis for the request is a high degree of reliability associated with the EFCVs and the low consequences from an EFCV failure. The analysis to support this conclusion was based on the Boiling Water Reactor Owners Group's (BWROG's) Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998, by General Electric Nuclear Energy. The BWROG provided clarifying information related to NRC staff questions in a letter dated January 6, 2000.

The supplemental letter dated January 26, 2000, provided clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination. In both the December 17, 1999, application and the January 26, 2000, supplement, the licensee bases its justification for the change on the topical report, as supplemented by the BWROG, and on plant-specific information provided.

2.0 BACKGROUND

EFCVs in reactor instrumentation lines are used in boiling water reactor (BWR) containments to limit the release of fluid from the reactor coolant system in the event of an instrument line break. Examples of EFCVs include reactor pressure vessel level/pressure instrument, main steam line flow instrument, recirculation pump suction pressure instrument, and reactor core isolation cooling steam line flow instrument. EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate following a loss-of-coolant accident (LOCA). The topical report states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design-basis LOCA would be of a sufficiently low probability to be outside of the design basis.

TS Surveillance Requirement (SR) 3.6.1.3.9 currently requires verification of the actuation (closing) capability of each reactor instrumentation line EFCV every 18 months. This requirement is typical for BWR TSs. The proposed change would relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every 18 months. The licensee proposes to test approximately 20 percent of the EFCVs every 18 months such that each EFCV will be tested at least once every 10 years (nominal). The proposed change is similar in principle to existing performance-based testing programs, such as inservice testing of snubbers and Option B of Appendix J to 10 CFR Part 50.

Licensees make changes to their TS Bases sections without the need for prior NRC review or approval, provided the Bases change does not involve an unreviewed safety question. Nevertheless, the licensee has included in its submittal, for information, a revised basis for SR 3.6.1.3.9. The revised basis states, in part:

The representative sample consists of an approximately equal number of EFCVs (about 15), from different plant locations and operating environments, such that each EFCV is tested at least once every 10 years.

and

The nominal 10-year interval is based on other performance testing. An EFCV failure will be evaluated per the Corrective Action and Maintenance Rule programs to determine if additional testing is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint....

3.0 EVALUATION

3.1 Systems Review

The topical report provides detailed information about EFCV surveillance testing at 12 BWR plants. Testing history indicates that there is a low failure rate in EFCV surveillance testing (see Section 3.2.1, below). At Fermi 2, there have been no failures in approximately 10 years of testing 93 valves. Thus, EFCVs have been very reliable performers, in general, and notably so at Fermi 2.

The dose consequences would be low if an EFCV failed to close upon an instrument line break during normal operation (see Section 3.2.2, below).

3.1.1 NRC Staff Generic Questions

The NRC staff raised some generic questions related to the topical report. These questions were first sent to IES Utilities, Inc. (IES), the licensee for the Duane Arnold plant, in letters dated September 27 and September 30, 1999. IES forwarded the questions to the BWROG for a generic response. The BWROG provided its answers to the questions to the NRC in a letter

dated January 6, 2000. The first three questions were related to the systems review and these will be addressed in the following subsections. The remaining three questions are related to the risk and radiological review. These questions will be addressed in Section 3.2.

3.1.1.1 Test Interval Increase

The topical report refers to Option B of Appendix J to 10 CFR Part 50 in its discussion of the extension of the test interval. The Commission revised Appendix J in 1995 by adding Option B, which provides a risk-informed, performance-based approach to leakage rate testing of containment isolation valves. The NRC staff also developed Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, as an acceptable method for implementing Option B. RG 1.163 states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B, with four exceptions that are described in the regulatory guide.

According to the NEI document, containment isolation valve test intervals may be increased to 5 years or three refueling outages if a valve has shown good performance (i.e., two consecutive successful tests), and, if certain other conditions are met, the interval may be increased to as much as 10 years. However, RG 1.163 took exception to these provisions of the NEI document, stating that test intervals should not exceed 5 years. RG 1.163 explained that this was because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical containment isolation valve performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in the NEI document to address these uncertainties.

The topical report states that the NEI document allows a 10-year test interval, and that RG 1.163 endorsed the NEI document, without mentioning the NRC staff's exception to the 10-year interval. The NRC staff asked IES (and through them, the BWROG) to justify its proposal for a 10-year testing interval. The BWROG responded that the topical report established its own basis for the testing relaxation (i.e., high reliability, low risk, and low radiological consequences). Nevertheless, the BWROG addressed the reasons stated in RG 1.163 for not accepting a 10-year interval as follows:

- Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified (i.e., in each plant's Updated Final Safety Analysis Report (UFSAR)). The dose consequences of the failure to isolate are acceptable (see Section 3.2.2, below).
- Repetitive/common-mode failures are not applicable, as evidenced by the low industry failure rate and more specifically by the BWROG topical report, Table 4-2, "EFCV Failure Rates by Manufacturer."
- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.

- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.
- There is no indeterminate time period involved with this proposed change. Every cycle (18 months for Fermi 2), a representative sample will be tested.

The NRC staff concludes that the BWROG's assessment is acceptable, except for the aging question that is addressed below in Section 3.2.1. RG 1.163 considered all varieties of containment isolation valves, from a fraction of an inch to several feet in diameter, carrying liquid or gas in a wide range of temperatures and pressures. The regulatory guide had to account for different types of valves (e.g., gate, globe, check) made of various materials, by different manufacturers, and with varying levels of safety significance. On the other hand, EFCVs in reactor instrumentation lines are a very specific, narrow class of valves. Their history and performance are well-documented. Based on their historically high reliability and their low risk significance and radiological consequences should they fail, the NRC staff concludes that the proposed extended test intervals are acceptable.

3.1.1.2 Failure Feedback Mechanism

The NRC staff pointed out to the BWROG that, under Appendix J, Option B, testing programs, a valve that fails a test after having been put on an extended test interval must return to its original interval until it once again shows good performance (i.e., passes two consecutive tests). RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," related to risk-informed inservice testing, also specifies the need for a failure feedback mechanism. Topical Report B21-00658-01 has no specific failure feedback mechanism.

The BWROG responded that each licensee who adopts the reduced surveillance intervals recommended by the subject topical report should ensure an appropriate feedback mechanism is in place to respond to failure trends. This issue is also addressed in a proposed generic change to the TS Bases for the EFCVs. In its December 17, 1999, application, the licensee stated that its existing corrective action program, along with its maintenance rule (10 CFR 50.65) program, would provide appropriate actions to correct future valve failures (see a more detailed discussion in Section 3.2.1, below).

Thus, considering the historically high reliability of the EFCVs and their low risk significance and radiological consequences should they fail, the NRC staff concludes that the licensee's program for responding to future test failures is sufficient.

3.1.1.3 Technical Specification Level of Detail

The proposed TS states that "a representative sample" of EFCVs will be tested every 18 months. The "representative sample" is not defined in the TS itself. The proposed Bases say that the licensee will test about 15 of the valves each refueling outage such that each EFCV is tested at least once every 10 years. The NRC staff asked the BWROG to justify placing the specific requirements in the Bases, rather than in the proposed TS.

The BWROG replied that the term "representative sample," with an accompanying explanation in the TS Bases, is identical to current usage in the Standard Technical Specifications (STS), NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term "representative" in

SR 3.8.6.3 in reference to battery cell testing, and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. Therefore, the application of a "representative sample" for the EFCV testing SR, with its accompanying definition in the Bases, is consistent with the STS usage.

Based on its evaluation of the information provided, the NRC staff concludes that the proposed TS wording is acceptable.

3.2 Risk and Radiological Review

In Topical Report B21-00658-01, the BWROG provided: (1) an estimate of the steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close and (2) an assessment of the radiological consequence of such release. The NRC staff's evaluation of this topical report, as supplemented, and as it pertains to Fermi 2, is provided below.

With two exceptions, the instrument lines at Fermi 2 include a 1/4-in flow restriction orifice to limit reactor water leakage in the event of a rupture. One exception is the jet pump flow instrument lines. These lines include a section of 0.25-inch piping from the jet pump taps to the reactor pressure vessel (RPV) nozzles, limiting flow in the same way as the orifices. The other exception is the feedwater pressure sensing lines. These lines tap into the feedwater lines outside of containment and, therefore, the inboard isolation check valves (B2100F010A/B) serve the function of the restricting orifices. As discussed in Section 3.2.2 below, the previous evaluation of such an instrument line rupture in Fermi 2 UFSAR, Section 15.6.2, for which the EFCVs are designed to provide a mitigating function, do not credit the isolation of the line by the EFCVs. Thus, a failure of an EFCV is bounded by the previous evaluation of an instrument line rupture. The UFSAR analysis also showed that the resulting offsite doses would be well below regulatory limits.

The operational impact of an EFCV that is connected to the RPV boundary failing to close is based on the environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. However, the topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following instrument line break would be met. The separation of equipment in the reactor building is also expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. Nevertheless, the presence of an unisolated steam leak into the reactor building would require the licensee to shut down and depressurize the reactor to allow access to manually isolate the line.

The licensee's evaluation of the frequency of steam release caused by an instrument line break concurrent with an EFCV failing to close is reviewed in Section 3.2.1 of this safety evaluation. The assessment of the radiological consequences of such release is reviewed in Section 3.2.2.

3.2.1 Estimation of Release Frequency

In estimating the release frequency initiated by an instrument line break, the topical report considered two factors: (1) the instrument line break frequency and (2) the probability of EFCV failing to close.

3.2.1.1 Estimation of Pipe Break Frequency

In the topical report, the BWROG used pipe break data based on WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," published in 1974. The NRC staff questioned why more recent data had not been used. In its January 6, 2000, letter, the BWROG assumed an instrument line break frequency of 3.52E-05/year. This estimate was based on the Electric Power Research Institute's Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," dated July 1992, and corresponds to pipe sizes between ½ inch to 2 inches in diameter. The licensee considers these pipe sizes to represent the subject instrument line piping. Thus, for Fermi 2, whose total number of instrument line/EFCVs is 93, the total plant instrument line break frequency would be estimated at 3.27E-3/year. The NRC staff concludes that this pipe break frequency represents recent pipe failure data and that the application of this frequency to represent the instrument line failure frequency is acceptable.

3.2.1.2 Estimation of EFCV Failure Frequency

The probability of an EFCV failing to close (or EFCV unavailability) was estimated by the BWROG using the formula:

$$\bar{A} = \lambda * \theta / 2$$

Where:

- \bar{A} is the EFCV unavailability
- λ is the EFCV failure rate per year
- θ is the EFCV surveillance test interval in years

The EFCV failure frequency, λ , was estimated using the formula:

$$\lambda_u = \chi^2_{\alpha; 2r+2} / 2T$$

Where:

- λ_u is the upper limit failure rate per year
- T is the operating time in years
- r is the number of failures
- $\chi^2_{\alpha; 2r+2}$ is the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom at $\alpha = 0.05$ (0.95 confidence level)

The topical report established an upper limit EFCV failure rate based on 11 observed failures in about 12424.5 years of service for 12 BWR plants in the United States (Note: 12424.5 years was determined by multiplying the number of tested EFCVs with the time period during which the number of occurring failures was reported). For 11 observed EFCV failures, the BWROG estimated that the EFCV failure rate, λ , was 1.67E-7 failures per hour.

However, the above formula for EFCV unavailability, \bar{A} , assumes that the EFCV failure rate is constant over time. The NRC questioned how the analyses accounted for possible changes in failure rate with time. The BWROG, in its response to the NRC's question, reported that it was not currently aware of any study that explores the causes of EFCV failures, or changes in EFCV failure rate over time. Nevertheless, to account for the possibility that the failure rate for EFCV may change over time, potentially due to age-related factors, the BWROG conservatively assumed the EFCV failure rate to change by five-fold. The NRC staff concludes that the licensee's method to account for a potential change in EFCV failure rate is acceptable.

For 55 EFCV failures (five times the actual number of EFCV failures observed for 12 BWR plants), degrees of freedom ($2r + 2$, where r is the number of failures) is 112. Chi-squared values, $\chi^2_{\alpha, 2r+2}$, are not typically provided for degree of freedom values above 30 because a chi-squared distribution with degrees of freedom over 30 approximates the standard normal distribution. In such cases, χ^2 is approximated by:

$$\chi^2 = \frac{1}{2} (Z + (2n-1)^{1/2})^2$$

Where:

- Z is the corresponding standard deviation (or a z-score) for α -point of the standard normal distribution
- n is the degrees of freedom

Thus, for a 0.95 confidence level ($\alpha = 0.05$), Z is 1.645.

$$\chi^2 = \frac{1}{2} (1.645 + (2 * 112 - 1)^{1/2})^2 = 137.42$$

Therefore, the BWROG calculated the EFCV upper limit failure frequency to be:

$$\lambda_u = \chi^2 / 2T = 137.42 / (2 * 1.09E+8 \text{ hours}) = 6.30E-7 \text{ failures per hour}$$

The release frequency was then calculated by the formula:

$$\begin{aligned} \text{RF} &= I * \bar{A} \\ &= I * \lambda_u * \theta / 2 \end{aligned}$$

Where:

- RF is the release frequency
- I is the instrument line failure frequency (per year)
- \bar{A} is the EFCV unavailability (calculated by $\lambda * \theta / 2$)
- θ is the EFCV surveillance test interval

Using the surveillance interval for 18 months (current practice), the instrument line break frequency of $3.27E-3$ /year at Fermi 2, and total plant EFCV failure frequency of $6.30E-7$ /hour, the release frequency for Fermi would be estimated to be $1.36E-5$ /year. For a surveillance interval of 10 years, the release frequency would be about $9.02E-5$ /year, which depicts an increase of about $7.66E-5$ /year from that of the 18-month surveillance test interval. It represents the increase in the total plant release frequency for a random break of any of the 93 instrument lines in Fermi 2 and a concurrent failure of the line's EFCV to isolate the break by closing.

The NRC staff considers the estimated increase in release frequency, $7.66E-5/\text{year}$, to be low. This is based on a qualitative analysis that an instrument line break with a concurrent failure of EFCV to close is not a significant contributor to core damage accidents. In addition, this increase in release frequency is lower than the Fermi 2 medium or large-break LOCA frequency of $7.1E-4/\text{year}$ that has the potential to lead to a core damage accident, whereas the instrument line break concurrent with EFCV failing to close does not. Based on these factors, the NRC staff does not consider the estimated increase in release frequency to be significant.

The NRC staff questioned the statement in the first paragraph of Section 4.2 of the topical report that there had been a total of 9 failures in over 10,000 valve years of operation. The BWROG indicated that it would correct this statement to indicate that there had been 11 failures.

The NRC staff also questioned whether there was additional failure data for this type of valve in other applications, how future failure data would be shared, what performance criteria would be used, and how failures would be evaluated and addressed. In its January 6, 2000, letter, the BWROG stated that EFCVs are not typically used in other applications. Therefore, the topical report provides the available failure information. The licensee, in its December 17, 1999, application, also indicated that it does not use these valves in any other applications at Fermi 2. With respect to sharing future failure data, the BWROG letter stated that the normal mechanisms (e.g., the Equipment Performance and Information Exchange, the Licensee Event Reporting system) would be used. Also, for future failures, the licensee stated that any EFCV failure would be evaluated under the Fermi 2 corrective action program and that its 10 CFR 50.65 maintenance rule program would be revised to include a specific EFCV performance criterion of less than or equal to one failure per year on a 3-year rolling average.

The NRC staff concludes that the method used by the BWROG for assessing the impact of EFCV surveillance test interval increase to 10 years (along with an assumed five-fold increase in the EFCV failure rate) is acceptable. The staff notes that the use of observed industry data for instrument line break and EFCV failures is sound, that the method of estimating the EFCV unavailability is consistent with industry practice, and that accounting for a potentially unknown change in the valve's failure rate is prudent. Finally, the NRC staff considers the licensee's commitment to monitor and evaluate EFCV failure rates under its corrective action and maintenance rule programs is both prudent and necessary.

3.2.2 Radiological Consequences

The licensee noted that it had previously evaluated the radiological consequences of an unisolable rupture of such an instrument line in Fermi 2 UFSAR, Section 15.6.2. This evaluation assumed a continuous discharge of reactor water through an instrument line with a 1/4-inch orifice for the duration of the detection and cooldown sequence. The assumptions for the accident evaluation do not change as a result of the proposed TS change, and the evaluation in Fermi 2 UFSAR, Section 15.6.2, remains acceptable. Therefore, the NRC staff finds acceptable the licensee's determination that the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

4.0 EVALUATION CONCLUSION

Increasing the EFCV surveillance test intervals to 10 years and assuming a five-fold increase in the EFCV failure rate results in a release frequency of about $9.02E-5$ /year. This represents an increase of about $7.66E-5$ /year from the current release frequency estimate (for an 18-month surveillance test interval) of about $1.36E-5$ /year. The NRC staff considers this estimate to be sufficiently low, especially since the consequence of such an accident is unlikely to lead to core damage. The staff also concludes that the consequences of the steam release from the depicted events is bounded by an existing UFSAR analysis. Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with unlikely impact on core damage, and negligible consequence of a release in the reactor building, the NRC staff concludes that the increase in risk associated with the licensee's request for relaxation of EFCV surveillance testing is low and the proposed change is acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a 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 (65 FR 4270). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Principal Contributor: A. Kugler

Date: March 14, 2000