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Detroit Edison

10CFR50.92
10CFR50.55a(a)(3)(i)

December 17, 1999
NRC-99-0101

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

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Proposed Technical Specification Change (License Amendment) to Relax Surveillance Testing Requirements for Excess Flow Check Valves and Submittal of Pertinent IST Relief Request

Pursuant to 10CFR50.90 and 10CFR50.55a(a)(3)(i), Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License NPF-43, Appendix A, Technical Specifications (TS) and requests the approval of Inservice Testing (IST) relief request number VRR-011. The proposed changes will modify TS Surveillance Requirement (SR) 3.6.1.3.9 to relax the SR frequency by allowing a representative sample of Excess Flow Check Valves (EFCVs) to be tested every 18 months, such that each EFCV will be tested at least once every ten years. The SR reflected in the current Improved Technical Specifications requires testing all EFCVs every 18 months. The IST relief request is being submitted to modify the IST program to be consistent with the proposed TS change.

The basis for this TS amendment is consistent with that described in the Boiling Water Reactor Owners' Group (BWROG) Report, B21-00658-01, dated November 1998. This report was submitted to the NRC with Duane Arnold Energy Center (Docket No. 50-331) proposed TS amendment, as a lead BWR plant, on April 12, 1999. Generic responses to the NRC staff questions posed to the lead plant are being submitted separately to the NRC by the BWROG. Additionally, Technical Specification Task Force (TSTF) Generic Traveler Number 334 was submitted to the NRC for approval.

A001

Enclosure 1 provides a description and evaluation of the proposed TS changes. Enclosure 2 provides an analysis of the issue of significant hazards consideration using the standards of 10CFR50.92. Enclosure 3 provides marked up pages of the current TS and Bases to show the proposed changes and a typed version of the affected TS and Bases pages with the proposed changes incorporated. Enclosure 4 provides IST relief request number VRR-011 for the second 120-month interval for NRC approval. Upon approval of this TS amendment and relief request VRR-011, refueling outage justification number ROJ-005, in the second 120-month interval IST program revision, will be replaced with relief request VRR-011.

Detroit Edison has reviewed the proposed TS changes against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor significantly change the types or significantly increase the amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed TS change meet the criteria provided in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

Detroit Edison requests that the NRC approves and issues the TS amendment and approves the relief request by March 31, 2000, with a 30-day implementation time. The proposed amendment is needed to minimize personnel radiation exposure during the upcoming seventh refueling outage scheduled to start on March 31, 2000.

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

Sincerely,

Enclosures

cc: A. J. Kugler
A. Vegel
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, DOUGLAS R. GIPSON, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

DOUGLAS R. GIPSON
Senior Vice President, Nuclear Generation

On this 17th day of December, 1999 before me personally appeared Douglas R. Gipson, being first duly sworn and says that he executed the foregoing as his free act and deed.

Notary Public

ROSALIE A. ARMETTA
Notary Public, Monroe County, MI
My Commission Expires Oct 11, 2003

ENCLOSURE 1

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

REQUEST TO REVISE TECHNICAL SPECIFICATIONS:

REVISION OF SURVEILLANCE REQUIREMENT FOR THE EXCESS FLOW CHECK VALVES

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGES

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGES

DESCRIPTION

Fermi 2 Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.9 currently requires verification of the actuation capability of each reactor instrumentation line Excess Flow Check Valve (EFCV) every 18 months. This SR demonstrates that each reactor instrumentation line EFCV is OPERABLE by verifying that the valve restricts flow on a simulated instrument line break downstream of the valve. The 18 month 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. Since testing requires the reactor to be pressurized to near normal operating pressure, this SR is normally performed during the reactor pressure vessel system leakage test, which is performed near the end of each refueling outage. EFCVs are tested by opening a downstream test drain valve from each EFCV and verifying proper operation.

All Instrument lines connected to the Reactor Coolant Pressure Boundary (RCPB) are equipped with a 0.25-inch flow-restricting orifice located as close as practical to the point of connection to the RCPB except for the jet pump flow instrument lines and the feedwater pressure-sensing lines. The jet pump lines are 0.25-inch in diameter from the jet pump taps to the RPV nozzles, and the feedwater pressure-sensing lines tap into the piping outside containment; therefore, the inboard isolation check valves (B2100F010A/B), located inside the containment, serve the function of the restricting orifices. Additionally, the main body orifice of EFCVs at Fermi 2 is 0.25-inch in diameter; therefore, it acts as another restricting orifice. A manual shutoff valve is located outside the containment and is located as close as practical to the containment wall or pipe (in the case of feedwater lines). The EFCV is located immediately downstream of the manual valve. This design and installation follows the guidance of Regulatory Guide 1.11. EFCVs at Fermi 2 are of the same size, make and model.

The proposed change is to relax the surveillance requirement frequency by allowing a representative sample of EFCVs to be tested every 18 months, such that each EFCV will be tested at least once every ten years. The proposed change is being requested to minimize personnel radiation exposure during refueling outages, cut down on outage critical path time and increase the availability of instrumentation during outages without significantly impacting the risk to the general public.

The Boiling Water Reactor Owners' Group (BWROG) has issued a report that provides a basis for this request. This report (B21-00658-01, dated November 1998) provides justification for the relaxation in the SR frequency as described above. The report demonstrates the high degree of

EFCVs reliability and the low consequences of an EFCV failure. A similar TS amendment has been submitted for the Duane Arnold Energy Center (Docket No. 50-331) on April 12, 1999.

Reliability data shown in the BWROG report documents no EFCV failures at Fermi 2. Any future EFCV failure would be evaluated per the Fermi 2 Corrective Action program. Additionally, as part of the implementation of this TS amendment, the 10CFR50.65 Maintenance Rule program will be revised to include a specific EFCVs performance acceptance criteria of less than or equal to one failure per year on a three-year rolling average.

EFCVs are included in the Fermi 2 Inservice Testing (IST) program. These containment isolation valves are not subject to 10CFR50, Appendix J, Type C testing. For the second 120-month interval, which will start on February 17, 2000, Refueling Outage Justification number ROJ-005 is included in the program revision to justify testing these valves at each refueling outage instead of the quarterly test requirement for check valves per the ASME Code. A new relief request number VRR-011 is included with this submittal for NRC approval. This proposed relief request is being submitted to modify the IST program requirements to be consistent with the proposed TS amendment. As part of the implementation of the proposed TS change, ROJ-005 will be superseded and replaced with VRR-011 for the second interval IST program.

EVALUATION OF THE PROPOSED CHANGES

The proposed changes to the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.9 will relax the SR frequency by allowing a representative sample of EFCVs to be tested every 18 months, such that all EFCVs will be tested at least once every ten years. This evaluation discusses the basis for the requested change.

Industry experience with EFCVs indicate that they have very low failure rates. There have been no failures associated with EFCV isolation testing at Fermi 2 (zero failures in 837 tests to date). There are no other valves similar to EFCVs at Fermi 2. The high reliability of these valves and the low risk significance associated with an EFCV failure to isolate an instrument line break are the primary bases for this change as documented in the BWROG report mentioned above. The report indicates that many reported test failures at other plants were related to test methodologies and not actual valve failures. As stated previously, the instrument lines at Fermi 2 include a flow-restricting orifice (or a 0.25-inch diameter line) upstream of each EFCV to limit reactor water leakage in the event of a rupture. The exception is the two feedwater pressure-sensing lines that tap into the feedwater lines outside of containment between the inboard and outboard containment isolation valves. In this configuration, the inboard isolation valves serve the function of the restricting orifices.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Subsection 15.6.2. The evaluation assumed the EFCV fails to isolate the break. Leakage from the break upstream of the excess-flow check valve is minimized by the line size or the restricting orifice in the line. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are substantially below the guidelines of 10CFR100. Therefore, a failure of an EFCV, though not expected as a result of this TS change, is bounded by the previous evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by this proposed change.

The reduced testing associated with this proposed change will result in an increase in the availability of the instrumentation during the outages, a saving in outage critical-path time and cost, and dose savings without significantly impacting the health and safety of the general public.

ENCLOSURE 2

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

REQUEST TO REVISE TECHNICAL SPECIFICATIONS:

REVISION OF SURVEILLANCE REQUIREMENT FOR THE EXCESS FLOW CHECK VALVES

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

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

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

   The current SR frequency requires each reactor instrumentation line EFCV to be tested every 18 months. The EFCVs at Fermi 2 are designed to close automatically in the event of a line break downstream of the valve. Indicating lights on a control room panel monitor EFCV positions. These valves may be reopened by actuation of a solenoid valve, which is operated from a local control panel. EFCVs at Fermi 2 are designed and installed following the guidance of Regulatory Guide 1.11. This proposed change allows a reduced number of EFCVs to be tested every 18 months. Industry operating experience, documented in BWROG Report B21-00658-01, concludes that a change in surveillance test frequency has a minimal impact on the reliability for these valves. A failure of an EFCV to isolate cannot initiate previously evaluated accidents; therefore, there can be no increase in the probability of occurrence of an accident as a result of this proposed change.

   Fermi 2 UFSAR, Subsection 15.6.2 evaluates an instrument line pipe break within secondary containment. The evaluation assumes that a small instrument line instantaneously and circumferentially breaks at a location where it may not be possible to isolate it and where immediate detection is not automatic or apparent. The evaluation concluded that pressurization of the secondary containment would not result from an instrument line break and a failure of the associated EFCV to isolate the ruptured line. The standby gas treatment system is not impaired by this event, and the calculated offsite exposure is substantially below the guidelines of 10CFR100. Additionally, coolant lost from such a break is inconsequential when compared to the makeup capabilities of the feedwater or RCIC system. The BWROG report concludes that the risk to the public with the extended testing interval is several orders of magnitudes below the general public annual exposure limits in 10CFR20.105.

   Although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by the analysis in the UFSAR. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the radiological consequences of an accident previously evaluated as a result of this change.

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

   This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in the BWROG report provides supporting evidence that the reduced testing frequency will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduction in test frequency is bounded by the evaluation of an instrument line pipe break described in Subsection 15.6.2 of the UFSAR. This change is not a physical alteration of the plant and will not alter the operation of the structures, systems and components as described in the UFSAR. Therefore, a new or different kind of accident will not be created.

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

   The consequences of a postulated instrument line pipe break have been evaluated in Subsection 15.6.2 of the UFSAR. The evaluation assumed the line instantaneously and circumferentially breaks at a location where it may not be possible to isolate it and that the EFCV fails to isolate the break. Therefore, any potential failure of an EFCV as a result of the reduced testing frequency is bounded by this evaluation and does not involve a significant reduction in the margin of safety.

ENCLOSURE 3

FERMI 2

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

REQUEST TO REVISE TECHNICAL SPECIFICATIONS

REVISION OF SURVEILLANCE REQUIREMENT FOR THE EXCESS FLOW CHECK VALVES

Attached is a mark-up of the existing Technical Specifications (TS) and TS Bases, indicating the proposed changes (Part 1), and a typed version of the TS and Bases incorporating the proposed changes (Part 2) with a list of included pages.

ENCLOSURE 3 - PART 1

PROPOSED TECHNICAL SPECIFICATIONS MARK-UP PAGE (INCLUDING TS BASES)

INCLUDED PAGES:

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 **AND** 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 ~~each~~ 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 each tested a representative sample of

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valve (EFCV) s ~~is~~ are OPERABLE by verifying that ~~the~~ valve restricts flow on a simulated instrument line break. Insert 1 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. 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. ~~Operating~~ Insert 2

Inserts to BASES for SR 3.6.1.3.9

Insert 1:

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.

Insert 2:

The nominal ten-year maximum limit is based on performance testing. An 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).

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~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 3.6.1.3.10

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 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.

BASES

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.

ENCLOSURE 3 - PART 2

PROPOSED TECHNICAL SPECIFICATIONS REVISED PAGE (INCLUDING TS BASES)

INCLUDED PAGES:

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 AND Once within 92 days after opening the valve
SR 3.6.1.3.7	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 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

SURVEILLANCE REQUIREMENTS (continued)

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

Secondary containment bypass leakage is also considered part of L_a.

SR 3.6.1.3.12

The analyses in References 1 and 4 are based on leakage that is less than the specified leakage rate. Leakage through all four main steam lines must be ≤ 100 scfh when tested at ≥ Pt (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_a (≥ 62.2 psig). Additionally, a combined leakage rate limit of ≤ 5 gpm when tested at 1.1 P_a (≥ 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.

ENCLOSURE 4

FERMI 2

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

REQUEST TO REVISE TECHNICAL SPECIFICATIONS

REVISION OF SURVEILLANCE REQUIREMENT FOR THE EXCESS FLOW CHECK VALVES

INSERVICE TESTING (IST) RELIEF REQUEST VRR-011 FOR THE SECOND 10-YEAR INTERVAL

VALVE RELIEF REQUEST VRR-011

SYSTEM: **NUCLEAR BOILER, REACTOR RECIRCULATION, REACTOR CORE ISOLATION COOLING, CORE SPRAY, HIGH PRESSURE COOLANT INJECTION, REACTOR WATER CLEANUP, AND REACTOR FEEDWATER**

VALVES:

Valve PIS No.	Code Class	Category	ISI Drawing
B21F501A	1	A/C	6M721-5808-1
B21F501B	1	A/C	6M721-5808-1
B21F501C	1	A/C	6M721-5808-1
B21F501D	1	A/C	6M721-5808-1
B21F502A	1	A/C	6M721-5808-1
B21F502B	1	A/C	6M721-5808-1
B21F502C	1	A/C	6M721-5808-1
B21F502D	1	A/C	6M721-5808-1
B21F503A	1	A/C	6M721-5808-1
B21F503B	1	A/C	6M721-5808-1
B21F503C	1	A/C	6M721-5808-1
B21F503D	1	A/C	6M721-5808-1
B21F504A	1	A/C	6M721-5808-1
B21F504B	1	A/C	6M721-5808-1
B21F504C	1	A/C	6M721-5808-1
B21F504D	1	A/C	6M721-5808-1
B21F506	1	A/C	6M721-5808-2
B21F507	1	A/C	6M721-5808-2
B21F508	1	A/C	6M721-5808-2
B21F509	1	A/C	6M721-5808-2
B21F510	1	A/C	6M721-5808-2
B21F511	1	A/C	6M721-5808-2
B21F512	1	A/C	6M721-5808-2
B21F513A	1	A/C	6M721-5808-2
B21F513B	1	A/C	6M721-5808-2
B21F513C	1	A/C	6M721-5808-2
B21F513D	1	A/C	6M721-5808-2
B21F514A	1	A/C	6M721-5808-2
B21F514B	1	A/C	6M721-5808-2
B21F514C	1	A/C	6M721-5808-2
B21F514D	1	A/C	6M721-5808-2
B21F515A	1	A/C	6M721-5808-2

Valve PIS No.	Code Class	Category	ISI Drawing
B21F515B	1	A/C	6M721-5808-2
B21F515C	1	A/C	6M721-5808-2
B21F515D	1	A/C	6M721-5808-2
B21F515E	1	A/C	6M721-5808-2
B21F515F	1	A/C	6M721-5808-2
B21F515G	1	A/C	6M721-5808-2
B21F515H	1	A/C	6M721-5808-2
B21F515L	1	A/C	6M721-5808-2
B21F515M	1	A/C	6M721-5808-2
B21F515N	1	A/C	6M721-5808-2
B21F515P	1	A/C	6M721-5808-2
B21F515R	1	A/C	6M721-5808-2
B21F515S	1	A/C	6M721-5808-2
B21F515T	1	A/C	6M721-5808-2
B21F515U	1	A/C	6M721-5808-2
B21F516A	1	A/C	6M721-5808-2
B21F516B	1	A/C	6M721-5808-2
B21F516C	1	A/C	6M721-5808-2
B21F517A	1	A/C	6M721-5808-2
B21F517B	1	A/C	6M721-5808-2
B21F517C	1	A/C	6M721-5808-2
B21F517D	1	A/C	6M721-5808-2
B31F501A	1	A/C	6M721-5809
B31F501B	1	A/C	6M721-5809
B31F501C	1	A/C	6M721-5809
B31F501D	1	A/C	6M721-5809
B31F502A	1	A/C	6M721-5809
B31F502B	1	A/C	6M721-5809
B31F502C	1	A/C	6M721-5809
B31F502D	1	A/C	6M721-5809
B31F503A	1	A/C	6M721-5809
B31F503B	1	A/C	6M721-5809
B31F504A	1	A/C	6M721-5809
B31F504B	1	A/C	6M721-5809
B31F505A	1	A/C	6M721-5809
B31F505B	1	A/C	6M721-5809
B31F506A	1	A/C	6M721-5809
B31F506B	1	A/C	6M721-5809
B31F510A	1	A/C	6M721-5809

Valve PIS No.	Code Class	Category	ISI Drawing
B31F510B	1	A/C	6M721-5809
B31F511A	1	A/C	6M721-5809
B31F511B	1	A/C	6M721-5809
B31F512A	1	A/C	6M721-5809
B31F512B	1	A/C	6M721-5809
B31F515A	1	A/C	6M721-5809
B31F515B	1	A/C	6M721-5809
B31F516A	1	A/C	6M721-5809
B31F516B	1	A/C	6M721-5809
E21F500A	1	A/C	6M721-5814
E21F500B	1	A/C	6M721-5814
E41F500	1	A/C	6M721-5815
E41F501	1	A/C	6M721-5815
E41F502	1	A/C	6M721-5815
E41F503	1	A/C	6M721-5815
E51F503	1	A/C	6M721-5816
E51F504	1	A/C	6M721-5816
E51F505	1	A/C	6M721-5816
E51F506	1	A/C	6M721-5816
G33F583	1	A/C	6M721-5818
N21F539A	1	A/C	6M721-5821
N21F539B	1	A/C	6M721-5821

FUNCTIONS:
Excess flow check valves are provided in each instrument process line that is part of the reactor coolant pressure boundary. The excess flow check valve is designed so that it will not close accidentally during normal operation, will close if a rupture of the instrument line occurs downstream of the valve, can be reopened when appropriate after closure from a local panel, and has its position indicated in the control room.

As detailed in the Fermi 2 UFSAR, Detroit Edison has incorporated into the design of each excess flow check valve source line the equivalent of a 0.25-inch restricting orifice. This was done by either the installation of a 0.25-inch orifice, the tap size of the source line being a 0.25-inch or in the case of the Feedwater pressure-sensing lines, taking credit for an inboard containment isolation valve. Additionally, the design of each excess flow check valve contains an internal 0.25-inch main body orifice. The restrictions in the source lines of the excess flow check valves limit leakage, in case of a failure to close, to a level where the integrity and functional performance of secondary containment and associated safety systems are maintained. The coolant loss is well within the capabilities of the reactor coolant makeup system, and the potential offsite exposure is substantially below the guidelines of 10CFR100.

Additionally, the design and installation of the excess flow check valves at Fermi 2 follow the guidance of Regulatory Guide 1.11.

OM-10 CODE REQUIREMETS FOR WHICH RELIEF IS REQUESTED:
OM-10 Section 4.3.2.1 requires that check valves, Category C valves, be exercised every 3 months to verify they fulfill their safety function.

BASIS FOR RELIEF:
Excess flow check valves are reliable devices, the major components are a poppet and spring. The spring holds the poppet open only under static conditions, such that the valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the valve. The resultant increase in flow imposes a differential pressure across the poppet which compresses the spring and closes off flow through the valve.

Excess flow check valves have been extremely reliable throughout the industry. Of the 837 tests performed in the first ten years of operation, no excess flow check valve isolation failures have been recorded (BWROG Report B21-00658-01). The Fermi 2 Technical Specifications detail what frequency is required to maintain a high degree of reliability and availability, and provide an acceptable level of quality and safety. Therefore, Detroit Edison requests relief pursuant to 10CFR50.55a(a)(3)(i) to test excess flow check valves at the frequency specified in Fermi 2 Technical Specifications Surveillance Requirements (SR) 3.6.1.3.9. As discussed in the Technical Specifications Bases for this SR, this test provides assurance that each valve restricts flow on a simulated instrument line break.

ALTERNATE TESTING:
Excess flow check valves will be tested at the frequency specified in Technical Specifications Surveillance Requirement 3.6.1.3.9.