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Nuclear

10 CFR 50.90

RS-03-106

June 10, 2003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Subject: LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Supplement to Request for Amendment to Technical Specifications
Surveillance Requirement 3.6.1.3.8

Reference: (1) Letter from K. Jury (EGC) to the NRC, "Request for Amendment to Technical Specifications Surveillance Requirement 3.6.1.3.8," dated April 18, 2003.

In accordance with 10 CFR 50.90, Exelon Generation Company (EGC), LLC, in Reference 1 requested an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18. Specifically, the proposed change would modify TS Surveillance Requirement (SR) 3.6.1.3.8 to identify that the specified testing requirement is applicable to reactor instrumentation lines. The proposed change was consistent with the SR wording specified in NUREG -1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 2, dated June 2001. Based on recent discussions with the NRC, EGC revised the supporting documentation and this supplemental submittal replaces the request contained in Reference 1.

The information supporting the proposed TS change is subdivided as follows.

- Attachment 1 is the notarized affidavit.
- Attachment 2 provides our evaluation supporting the proposed change.
- Attachment 3 contains a copy of the marked up TS page.
- Attachment 4 provides the retyped TS page.

The proposed TS change has been reviewed by the LaSalle County Station Plant Operations Review Committee (PORC) and approved by the Nuclear Safety Review Board (NSRB) in accordance with the Quality Assurance Program.

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EGC is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official. We request approval of the proposed change by January 1, 2004 to support the planning of testing during the scheduled Unit 1 refueling outage in February 2004.

Should you have any questions concerning this submittal, please contact Mr. T. W. Simpkin at (630) 657-2821.

Sincerely,


for Keith R. Jury
Director-Licensing
Mid-West Regional Operating Group

Attachments:

- Attachment 1. Affidavit
- Attachment 2. Evaluation of Proposed Change
- Attachment 3. Markup of Proposed Technical Specification Page Change
- Attachment 4. Retyped Page for Technical Specification Change

cc: Regional Administrator – NRC Region III
NRC Project Manager – NRC NRR
NRC Senior Resident Inspector – LaSalle County Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

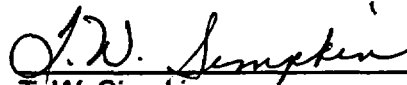
ATTACHMENT 1
Affidavit

STATE OF ILLINOIS)
COUNTY OF DUPAGE)
IN THE MATTER OF:)
EXELON GENERATION COMPANY (EGC), LLC) Docket Numbers
LASALLE COUNTY STATION - UNIT 1 and UNIT 2) 50-373 and 50-374

**SUBJECT: Supplement to Request for Amendment to Technical
Specifications Surveillance Requirement 3.6.1.3.8**


AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information, and belief.


T. W. Simpkin
Manager-Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and
for the State above named, this 10th day of
June, 2003




Notary Public

ATTACHMENT 2
Evaluation of Proposed Change
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- 1.0 INTRODUCTION
- 2.0 DESCRIPTION OF PROPOSED AMENDMENT
- 3.0 BACKGROUND
- 4.0 REGULATORY REQUIREMENTS & GUIDANCE
- 5.0 TECHNICAL ANALYSIS
- 6.0 REGULATORY ANALYSIS
- 7.0 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
- 8.0 ENVIRONMENTAL CONSIDERATION
- 9.0 PRECEDENT

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Evaluation of Proposed Change
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1.0 INTRODUCTION

In accordance with 10 CFR 50.90, Exelon Generation Company (EGC), LLC, in Reference 1 requested an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18. Specifically, the proposed change would modify TS Surveillance Requirement (SR) 3.6.1.3.8 to identify that the specified testing requirement is applicable to reactor instrumentation lines. The proposed change was consistent with the SR wording specified in NUREG -1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 2, dated June 2001. Based on recent discussions with the NRC, EGC revised the supporting documentation and this supplemental submittal replaces the request contained in Reference 1.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed wording associated with the change is identified below in bold type.

SR 3.6.1.3.8 Verify each **reactor instrumentation line** EFCV actuates to the isolation position on an actual or simulated instrument line break signal.

3.0 BACKGROUND

Excess flow check valves (EFCVs) are used as a means of automatic isolation on all static instrument sensing lines that penetrate the drywell containment. EFCVs that are connected to the reactor coolant pressure boundary (RCPB) are classified as reactor instrumentation line EFCVs. The LaSalle County Station TS was modified in content and format to the latest guidance provided by NUREG-1433, Rev. 2 in March 2001. During the conversion process the reference to testing of the EFCV was incorrectly changed to omit the clarifying statement that was contained in the existing SR (i.e., SR 4.6.3.4) and NUREG-1433, that the EFCV SR testing was specifically associated with the reactor instrument line EFCVs.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

10 CFR 50.36(c)(2)(ii)(c), "Criterion 3," requires that a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier be included in the TS.

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5.0 TECHNICAL ANALYSIS

5.1 Design Bases

NUREG-1433, Rev. 2, provided licensees with the latest NRC recommended content and format for TS. The NUREG-1433 SR for testing EFCVs, SR 3.6.1.3.10, specifies that this testing is associated with reactor instrumentation line EFCVs.

The proposed change will incorporate the wording from NUREG-1433 into LaSalle County Station SR 3.6.1.3.8 to specify that the scope of TS required testing is EFCVs that are directly connected to the RCPB (i.e., reactor instrumentation line EFCVs).

5.2 Risk Information

This submittal is not based on risk informed decision making.

6.0 REGULATORY ANALYSIS

The proposed change to SR 3.6.1.3.8 will provide the testing requirements for EFCVs that are associated with instrument lines that are connected to the RCPB and penetrate the drywell. The function of reactor instrumentation line EFCVs, in combination with other accident mitigation features, is to limit fission product release. Therefore, the testing of reactor instrumentation line EFCVs must be included in LaSalle County Station TS in accordance with 10 CFR 50.36(c)(2)(ii)(c).

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

EGC has evaluated the proposed change to the TS for LaSalle County Station, Unit 1 and Unit 2, and has determined that the proposed change does not involve a significant hazards consideration and is providing the following information to support a finding of no significant hazards consideration.

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change to the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 will incorporate into the SR wording specified in NUREG -1433, "Standard Technical Specifications General Electric Plants,

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BWR/4," Revision 2, dated June 2001. The proposed change will specify that the testing required by SR 3.6.1.3.8 is applicable to reactor instrumentation line excess flow check valves (EFCVs). The performance of TS surveillance testing is not a precursor to any accident previously evaluated. Thus, the proposed change does not have any affect on the probability of an accident previously evaluated.

The function of reactor instrumentation line EFCVs, in combination with other accident mitigation features, is to limit fission product release. The surveillance testing specified in SR 3.6.1.3.8 will provide assurance that the reactor instrumentation line EFCVs will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

Does the change involve a significant reduction in a margin of safety?

Response: No

NUREG-1433, Rev. 2, provided licensees with the latest NRC recommended content and format for TS. The NUREG-1433 SR for testing EFCVs, SR 3.6.1.3.10, specifies that this testing is associated with reactor instrumentation line EFCVs. The LaSalle County Station TS was modified in content and format to the latest guidance provided by NUREG-1433, Rev. 2 in March 2001. During the conversion process the reference to testing of the EFCV was incorrectly changed to omit the clarifying statement that was contained in the existing SR (i.e., SR 4.6.3.4) and NUREG-1433, that the EFCV SR testing was specifically associated with the reactor instrument line EFCVs. The proposed change will incorporate the wording from NUREG-1433 into LaSalle County Station SR 3.6.1.3.8.

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Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed amendment incorporates into the LaSalle County Station a change to SR 3.6.1.3.8 that is consistent with the wording and intent of NUREG-1433, Rev. 2.

ATTACHMENT 3

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGE

Revised TS Page

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6 Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7 Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
<div data-bbox="581 741 1105 804" style="border: 1px solid black; padding: 2px; display: inline-block;">REACTOR INSTRUMENT LINE</div> SR 3.6.1.3.8 Verify each VEFV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9 Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10 Verify leakage rate through any one main steam line is ≤ 100 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11 Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

ATTACHMENT 4

**RETYPE PAGE
FOR
TECHNICAL SPECIFICATION CHANGE**

Retyped TS Page

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrument line EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through any one main steam line is ≤ 100 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program