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10 CFR 50.90

RS-03-036

February 14, 2003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for Technical Specifications Surveillance Requirement 3.6.1.3.8 Change Related to Excess Flow Check Valve Testing

References:
(1) General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000

(2) NRC Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," March 14, 2000

(3) Letter from D. M. Skay (U. S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Approval to Implement a Check Valve Inservice Testing Program Using ASME OM Code-1995 Edition, OMA-1996 Addenda at the Commonwealth Edison Company Nuclear Stations," June 7, 2000

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-19 and DPR-25, for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed change revises TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.8 to require that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) be tested every

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24 months, such that each EFCV will be tested nominally at least once every 10 years. Currently, TS SR 3.6.1.3.8 requires testing of each reactor instrumentation line EFCV on a 24-month frequency. The proposed change in EFCV testing is consistent with Reference 1. In Reference 2, the NRC approved the generic applicability of Reference 1.

The proposed TS change is being requested to minimize personnel radiation exposure during refueling outages and to increase the availability of instrumentation during outages. The next DNPS refuel outage is currently scheduled to commence on November 4, 2003. To support incorporation of the EFCV testing relaxation into the schedule for the upcoming DNPS Unit 2 refuel outage (D2R18), EGC requests approval of the proposed amendments by October 24, 2003. Once approved, the amendment will be implemented within 30 days.

This request is subdivided as follows.

1. Attachment A gives a description and safety analysis of the proposed change.

2. Attachment B-1 provides the marked-up TS and Bases pages indicating the proposed change for DNPS. The marked-up Bases pages are provided for review purposes only, and do not require NRC approval. Attachment B-2 provides revised DNPS TS pages incorporating the proposed change.

3. Attachment C-1 provides the marked-up TS and Bases pages indicating the proposed change for QCNPS. The marked-up Bases pages are provided for review purposes only, and do not require NRC approval. Attachment C-2 provides revised QCNPS TS pages incorporating the proposed change.

4. Attachment D describes the evaluation performed using the criteria in 10 CFR 50.91(a), "Notice for public comment," paragraph (1), which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92, "Issuance of amendment," paragraph (c).

5. Attachment E provides information supporting an environmental assessment.

In Reference 3, the NRC approved implementation of the check valve condition monitoring program for inservice testing program check valves for DNPS and QCNPS. The condition monitoring program allows flexibility in establishing the types of tests, examination, and preventive maintenance activities and their associated intervals, when justified based on the valve's performance and operating condition. Since EGC will implement the performance-based condition monitoring program for EFCVs, additional relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code is not needed to support the proposed relaxation in EFCV testing frequency.

This proposed TS change has been reviewed by the Plant Operations Review Committees at each of the stations and approved by the respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Keith R. Jury

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

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for Proposed Change
Attachment B-1: Marked-Up Technical Specifications and Bases Pages for Proposed Change, Dresden Nuclear Power Station, Units 2 and 3
Attachment B-2: Typed Page for Technical Specifications Change, Dresden Nuclear Power Station, Units 2 and 3
Attachment C-1: Marked-Up Technical Specifications and Bases Pages for Proposed Change, Quad Cities Nuclear Power Station, Units 1 and 2
Attachment C-2: Typed Page for Technical Specifications Change, Quad Cities Nuclear Power Station, Units 1 and 2
Attachment D: Information Supporting a Finding of No Significant Hazards Consideration
Attachment E: Information Supporting an Environmental Assessment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
EXELON GENERATION COMPANY, LLC	)	Docket Numbers
DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3	)	50-237 and 50-249
QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2	)	50-254 and 50-265

SUBJECT: Request for Technical Specifications Surveillance Requirement 3.6.1.3.8 Change Related to Excess Flow Check Valve Testing

AFFIDAVIT

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

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

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 14th day of February, 2003.

Timothy A. Byam
Notary Public

"OFFICIAL SEAL"
TIMOTHY A. BYAM
COMMISSION EXPIRES 12/04/05

Attachment A

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGE

A. SUMMARY OF THE PROPOSED CHANGE

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) proposes a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. Specifically, EGC proposes to revise TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.8 to require testing of "a representative sample" of reactor instrumentation line excess flow check valves (EFCVs) such that each EFCV will be tested nominally at least once every 10 years. Currently, SR 3.6.1.3.8 requires testing of each reactor instrumentation line EFCV on a 24-month frequency. This proposed change is similar to previous changes that resulted in performance-based testing programs, such as Inservice Testing of snubbers and Option B to 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The basis for this change is consistent with General Electric Nuclear Energy Licensing Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" (Reference 1), prepared for the Boiling Water Reactor Owners' Group (BWROG). The generic applicability of this topical report was approved by the NRC in a safety evaluation (SE) dated March 14, 2000 (Reference 2). This proposed change is also consistent with standard technical specification change traveler TSTF-334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2 (Reference 3). TSTF-334, Revision 2, was approved by the NRC on October 31, 2000 (Reference 4). Reference 1 provides justification for a relaxation in the SR frequency by demonstrating a high degree of reliability for the EFCVs through operating experience and the low consequences of an EFCV failure.

A complete description of the proposed change is given in Section E of this Attachment. Attachments B-1 and C-1 provide the marked-up TS and Bases pages for DNPS and QCNPS, respectively, indicating the proposed change. Attachments B-2 and C-2 provide the typed TS page for DNPS and QCNPS, respectively.

B. DESCRIPTION OF THE CURRENT REQUIREMENT

The DNPS and QCNPS TS currently require the performance of surveillance tests on each reactor instrumentation line EFCV every 24 months. TS SR 3.6.1.3.8, requires a demonstration that each reactor instrumentation line EFCV is operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break signal. This SR provides assurance that the reactor instrumentation line EFCVs will perform as designed. SR 3.6.1.3.8 currently states:

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.

C. BASES FOR THE CURRENT REQUIREMENT

The function of PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated design basis accidents (DBAs) to within analyzed values. The operability requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the operability requirements provide assurance that the primary containment function assumed in the safety analysis will be maintained. Check valves, such as EFCVs, and automatic valves designed to close without operator action following an accident, are considered active devices. EFCVs are used in instrument lines to isolate a ruptured instrument line. The EFCV closes as a result of high differential pressure in the instrument line.

The current 24-month surveillance frequency is based on testing during a plant outage to avoid a potential for an unplanned transient if the testing were performed with the reactor at power.

D. NEED FOR REVISION OF THE REQUIREMENT

The BWROG has developed a basis for relaxing the requirement to test each EFCV connected to the reactor coolant pressure boundary (RCPB) during each refueling outage. The requested change is consistent with Reference 1, as approved by the NRC in Reference 2. The NRC has approved similar changes for several other plants (References 5 through 9). The change proposed herein to the DNPS and QCNPS TS is consistent with these approved license amendments. The reduced testing associated with the proposed change will result in an increase in the availability of the instrumentation during outages and a savings in personnel dose, without impacting plant safety.

E. DESCRIPTION OF THE PROPOSED CHANGE

The proposed change revises SR 3.6.1.3.8 to state:

SR 3.6.1.3.8 Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.

The proposed TS change is reflected on a marked-up copy of the affected TS page for DNPS and QCNPS in Attachments B-1 and C-1, respectively. Marked-up Bases pages are also included for review purposes. Revised TS pages incorporating the proposed change are provided in Attachments B-2 and C-2 for DNPS and QCNPS, respectively. Following NRC approval of this request, EGC will revise the DNPS and QCNPS TS Bases, in accordance with the TS Bases Control Program of TS Section 5.5.10, "Technical Specifications (TS) Bases Control Program," to incorporate the changes.

The proposed change to the TS also requires a corresponding change to the applicable sections of the DNPS and QCNPS Inservice Testing (IST) Programs. The IST Programs would then require that every 24 months, a representative sample of reactor instrumentation

line EFCVs be tested to satisfy the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Following NRC approval of the proposed TS change, the required changes to the DNPS and QCNPS IST Programs will be done in accordance with the Condition Monitoring Process defined in Appendix II, "Check Valve Condition Monitoring Program," of the ASME OMa Code-1996. The NRC granted DNPS and QCNPS permission to utilize the ASME OM Code-1995 Edition, OMa-1996 Addenda, which included Appendix II, in Reference 10.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGE

The proposed change will increase the test interval of the EFCVs. Reference 1 compares this situation to Option B of 10 CFR 50, Appendix J. The NRC 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. As discussed in Reference 2, the NRC accepted the EFCV test interval extension, which may be as long as 10 years, based on the EFCVs historically high reliability, their low risk significance, and the low radiological consequences should they fail.

The NRC approved the generic use of Reference 1 if licensees perform the following functions.

1. Perform a plant-specific radiological dose assessment for an instrument line break.
2. Perform a plant-specific EFCV failure rate analysis.
3. Determine the plant-specific estimated release frequency initiated by an instrument line break.
4. Develop a plant-specific feedback mechanism and corrective action program to ensure EFCV performance.

Radiological Dose Assessment

DNPS and QCNPS each have 75 EFCVs per unit, installed in instrumentation lines connected to the RCPB, which serve as PCIVs. These EFCVs limit the release of inventory from the RCPB in the event of an instrument line break. The EFCVs and the associated instrument lines are considered extensions of the primary containment. The postulated break of an instrument line attached to the RCPB is discussed and evaluated in each station's Updated Final Safety Analysis Report (UFSAR), Section 15.6.2, "Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment." For both stations, the calculated potential offsite exposures for such instrument line breaks are well below the guidelines of 10 CFR 100, "Reactor Site Criteria." However, the instrument line break analysis does not credit closure of the associated EFCV. Therefore, the failure of an EFCV, though not expected as a result of this proposed change, is bounded by the evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by this proposed change.

EFCV Failure Rate Analysis

Reference 1 states in Section 2.2.4 that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break outside of primary containment coincident with a design basis Loss of Coolant Accident (LOCA) would be of sufficiently low probability to be outside of the design basis. Reference 1 Table 4-1, "EFCV Failure Rates," also provides detailed information about the results of EFCV testing at 12 Boiling Water Reactor (BWR) plants, including DNPS, Units 2 and 3. Reference 1 determined an upper limit EFCV failure rate based on 12,424.5 valve operating years (i.e., 1.09E+08 valve operating hours) with a plant average of 1035 valve operating years per plant. Considering the total number of EFCV failures (i.e., 11) out of 1.09E+08 valve operating hours for the 12 plants, Reference 1 concluded that EFCVs had a low failure rate (i.e., a nominal failure rate of 1.01E-07 failures per valve operating hour). Specifically for DNPS, Table 4.1 of Reference 1 specifies no test failures in 922 valve operating years (i.e., 8.07E+06 valve operating hours) for the total number of EFCVs tested (i.e., 150) in Units 2 and 3. In addition, there have been no additional EFCV failures at DNPS since test data was collected to support Reference 1. Therefore, EGC concludes that the EFCVs installed in DNPS are highly reliable and are bounded by the above failure rate specified in Reference 1.

EFCV testing data for QCPNS is not provided on Table 4.1 of Reference 1. Therefore, EGC reviewed the QCNPS surveillance test results since 1996 for reactor instrumentation line EFCVs. During this period, each Unit 1 EFCV was tested four times, and each Unit 2 EFCV was tested three times. A total of four EFCV failures occurred during this time period, two on Unit 1 and two on Unit 2. These tests cover a total of approximately 80,006 operating hours for both units (i.e., operating hours based on generator online hours). Total operating hours can be equated to EFCV operating hours by multiplying by the number of EFCVs per unit (i.e., 75 for QCNPS). Therefore, the QCNPS surveillance test results covered a period of approximately 6.00E+06 valve operating hours. These EFCV failures equate to an estimated nominal failure rate of 6.67E-07 per valve operating hour. Using the methodology described in Section 4.2 of Reference 1, an upper limit EFCV failure rate of 1.52E-06 failures per valve operating hour was calculated for QCNPS for the normal 24-month surveillance interval. The calculated upper limit failure rate, which is used in determining the estimated release frequency below, provides an estimate, with a 95% confidence level, of the reliability of QCNPS EFCVs based on testing experience. The QCNPS 24-month surveillance interval EFCV failure rate is higher than the industry composite upper limit failure rate specified in Reference 1 (i.e., 1.67E-07 failures per valve operating hour).

It should be noted that two of the four QCNPS EFCV failures specified above occurred during the first surveillance test performed on each unit during the 80,006 operating hours. The test methodology utilized for each unit's first surveillance test performed during the 80,006 operating hours differed from the test methodology utilized since then. The first surveillance test methodology required measurement of fluid flow through the tested EFCV with a quantitative acceptance criteria. If a tested EFCV failed to close within the quantitative fluid flow acceptance criteria value, the EFCV failed the surveillance test. This test methodology is more conservative than the test methodology specified in TS SR 3.6.1.3.8. SR 3.6.1.3.8 requires verification of a distinctive "click" when the tested EFCV

closes, without measurement of fluid flow through the EFCV. Since the first surveillance test for each unit during the 80,006 operating hours, QCNPS has changed the EFCV surveillance test methodology utilized to the test methodology specified in SR 3.6.1.3.8. Subsequent surveillance tests performed on each unit utilizing the test methodology specified in SR 3.6.1.3.8 recorded only two EFCV failures. Therefore, EGC concludes that if the current surveillance test methodology was utilized during the first surveillance test for each unit, fewer EFCV failures would have occurred, and the QCNPS EFCV failure rate would be closer to the industry composite failure rate specified in Reference 1. Although the previously determined QCNPS EFCV failure rate is greater than the industry composite failure rate, based on the change in QCNPS EFCV test methodology, EGC still considers the EFCVs installed in QCNPS to be highly reliable and comparable to the industry composite EFCVs.

Estimated Release Frequency Determination

In estimating the release frequency initiated by an instrument line break concurrent with an EFCV failure to close, two factors are considered: (1) the instrument line break frequency and (2) the probability of an EFCV failing to close. As noted in Section 3.2.1 of Reference 2, the BWROG assumed a single instrument line break frequency of 3.52E-05 per year. Thus, the product of this single instrument line break frequency and the total number of instrument lines with EFCVs results in a total instrument line break frequency. Since DNPS and QCNPS have the same number of RCPB EFCVs installed per unit (i.e., 75) the total instrument line break frequency is the same for each station. Total instrument line break frequency is calculated as follows:

- For each DNPS / QCNPS unit - (3.52E-05) X (75) = 2.64E-03 per year

In accordance with Reference 2, the estimated release frequency is the product of: (1) the total instrument line break frequency per year, (2) the total plant EFCV failure rate, and (3) the surveillance interval in years divided by 2. For DNPS, a total plant EFCV failure rate of 5.53E-03 per year as provided in Section 3.2.1 of Reference 2 was utilized since the DNPS EFCV failure rate is bounded by the industry composite rate specified on Table 4-1 of Reference 1. For QCNPS, the previously calculated total plant-specific upper limit EFCV failure rate of 1.52E-06 per hour (i.e., 1.34E-02 per year) is utilized since the QCNPS EFCV failure rate is not bounded by the industry composite rate. For the current surveillance interval of 24 months (i.e., 2 years), the estimated release frequency is calculated as follows:

- For each DNPS unit - (5.53E-03) X (2.64E-03) X (2/2) = 1.46E-05 per year
- For each QCNPS unit - (1.34E-02) X (2.64E-03) X (2/2) = 3.54E-05 per year

For the maximum surveillance-testing interval of 10 years, the estimated release frequency is:

- For each DNPS unit - (1.46E-05) X (10/2) = 7.30E-05 per year
- For each QCNPS unit - (3.54E-05) X (10/2) = 1.77E-04 per year

The estimated release frequencies for the 10-year surveillance testing interval represent an increases from the current 24-month testing interval of:

- For each DNPS unit - 5.84E-05 per year
- For each QCNPS unit - 1.42E-04 per year

The release frequency increases represent the increases in the total plant release frequencies for a random break of any of the total number of reactor instrument lines with EFCVs and a concurrent failure of the line's EFCV to isolate the break by closing. For DNPS, the value calculated is more conservative than the results of Reference 2 that concluded an increase in release frequency of 7.3E-05 per year was not significant. For QCNPS, the value calculated is nearly twice this Reference 2 release frequency increase. However, the QCNPS release frequency increase is lower than the release frequency increase calculated for Susquehanna Steam Electric Station, Units 1 and 2 (i.e., 2.02E-04 per year) as noted in Reference 8. The NRC approved Susquehanna's request for a 10-year EFCV surveillance interval based on the conclusion that Susquehanna's increase in the estimated EFCV release frequency was sufficiently low when considered in conjunction with the planned failure feedback and corrective actions as discussed in Reference 8. EGC's planned failure feedback and corrective actions, as described below, are consistent with Susquehanna's proposal, and provide for acceptance criteria that are more conservative with respect to Reference 2. Therefore, EGC concludes that the estimated release frequency, and increase in estimated release frequency, for a 10-year surveillance interval at both DNPS and QCNPS are sufficiently low.

Feedback Mechanism and Corrective Action Program

The reviewer's note, associated with TSTF-334, addresses the requirements for adopting the EFCV relaxation, including the selection of performance criteria and basis to ensure that the licensee's corrective action program can provide meaningful feedback for appropriate corrective actions. Any EFCV failures that may occur will be documented in the EGC Corrective Action Program as a surveillance test failure. The check valve Condition Monitoring Program will ensure that the failure will be evaluated to identify common failure mode, industry experience, and to review for similar component failure history.

EFCV testing will follow the same 10-year interval as the IST Program. DNPS and QCNPS operate on a two-year operating cycle (i.e., a refueling cycle approximately every 24 months). The EFCVs will be grouped in accordance with the IST Program's Condition Monitoring Program. Condition monitoring plans will be re-evaluated every two years, which includes review of test history, effectiveness of corrective actions, and consideration of appropriateness of current test frequencies. The test frequency must be periodically justified and approved by an expert panel as described in the Condition Monitoring Program. This approach will require that a continuing review of the failures be performed to assess performance trends.

The initial plan is to group the EFCVs into five groups with approximately 20 percent of the valves in each group. Each refuel outage one group (i.e., approximately 20%) will be tested. Testing procedures will be populated with valves from the five groups such that, during each outage, valves subject to steam conditions and valves subject to liquid conditions will be

tested to ensure a representative cross section of valve applications and service conditions are tested each refuel outage.

If any one EFCV in a group fails to check flow as a result of the test methodology or valve failure, testing of an additional group will be performed prior to restart. If any one EFCV fails to check flow in the additional group, the sample would expand to 100 percent of the EFCVs for the unit being tested prior to restart. Any valve that fails to check flow will be tested again in the next refueling outage, in addition to the normally required test population. This expansion is conservative, but reasonable, based on the historical performance of EFCVs at DNPS and QCNPS.

After NRC approval of the proposed license amendment, EGC will incorporate the performance criteria discussed above into the EFCV test procedures and into IST Program documents. The increased use of the Corrective Action and Condition Monitoring Programs will ensure that a heightened level of attention will be given to valve failures and that corrective actions are established to further improve test performance. Based on the above, the proposed change for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, meets the overall requirements to implement TSTF-334, Revision 2 (Reference 3).

G. IMPACT ON PREVIOUS SUBMITTALS

EGC has reviewed the proposed change for impact on any previous submittals, and has determined that there is no impact on any outstanding license amendment requests.

H. SCHEDULE REQUIREMENTS

We request approval of these proposed changes by October 24, 2003. Once approved, the amendment will be implemented within 30 days.

I. REFERENCES

1. General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000

2. NRC Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," (TAC Nos. MA7884 and M84809), dated March 14, 2000

3. Standard Technical Specification Change Traveler TSTF-334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2

4. NRC letter to Mr. Anthony R. Pietrangelo, Director, Risk & Performance Regulation, Nuclear Generation Division, Nuclear Energy Institute (NEI), concerning approval of sixteen NEI Technical Specification Task Force recommended changes to Standard Technical Specification NUREGs, dated October 31, 2000

5. NRC letter "Fermi 2 - Issuance of Amendment Re: Revised Excess Flow Check Valve Surveillance Requirements (TAC No. MA7373)," dated March 2000

6. NRC letter "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Excess Flow Check Valve Surveillance Intervals (TAC Nos. MA6407 and MA6409)," dated January 29, 2001

7. NRC letter "Limerick Generating Station, Units 1 and 2 - Issuance of Amendment Re: Revised Excess Flow Check Valve Surveillance Requirements (TAC Nos. MA9927 and MA9928)," dated February 23, 2001

8. NRC letter "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Regarding Relaxation of Excess Flow Check Valve Surveillance Requirements (TAC Nos. MB0424 and MB0427)," dated April 11, 2001

9. NRC letter "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Regarding Surveillance Testing of Excess Flow Check Valves (TAC Nos. MB1048 and MB1049)," dated October 4, 2001

10. NRC letter to Mr. Oliver D. Kingsley (Commonwealth Edison Company), "Approval to Implement a Check Valve Inservice Testing Program Using ASME OM Code-1995 Edition, OMa-1996 Addenda at the Commonwealth Edison Company Nuclear Stations (TAC Nos. MA8703, MA8704, MA8715, MA8716, MA8717, MA8718, MA8803, MA8804, MA8733, and MA8734)," dated June 7, 2000

Attachment B-1

MARKED-UP TECHNICAL SPECIFICATIONS AND BASES PAGES FOR PROPOSED CHANGE

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
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~~ A REPRESENTATIVE Sample of reactor instrumentation line ~~EFCV~~ EFCVs actuate~~s~~ 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.6

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 and transient analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

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 LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 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 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

A Representative Sample of

SR 3.6.1.3.8

S S Are

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valve (EFCV) ~~is~~ OPERABLE by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break condition. This test is performed by blowing down the instrument line during an inservice leak or hydrostatic test and verifying a distinctive "click" when the poppet valve seats or a quick reduction in flow. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned

INSERT 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.8 (continued)

transient if the Surveillance were performed with the reactor at power. ~~Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

INSERT 2

SR 3.6.1.3.9

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. The Frequency of 24 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). Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges must be followed.

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. The leakage rate of each main steam isolation valve path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves). If both isolation valves in the penetration are closed the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with the Primary Containment Leakage Rate Testing Program). The combined leakage through all MSIV leakage paths must be $\leq$ 46 scfh when tested at $\geq$ 25 psig. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

MSIV leakage is considered part of L_a.

(continued)

BASES (continued)

REFERENCES

1. Technical Requirements Manual.
2. UFSAR, Section 15.6.5.
3. UFSAR, Section 15.6.4.
4. UFSAR, Section 15.2.4.
5. UFSAR, Section 6.2.4.1.

INSERT 3

DRESDEN 2 AND 3

Insert 1:

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time.

Insert 2:

The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 6). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval 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.

Insert 3:

6. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000

Attachment B-2

TYPED PAGE FOR TECHNICAL SPECIFICATIONS CHANGE

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
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 a representative sample of reactor instrumentation line EFCVs actuate 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.6

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 and transient analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

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 LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 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 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break condition. This test is performed by blowing down the instrument line during an inservice leak or hydrostatic test and verifying a distinctive "click" when the poppet valve seats or a quick reduction in flow. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.8 (continued)

every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 24 month Frequency is based on the need to perform 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 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 6). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval 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.

SR 3.6.1.3.9

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. The Frequency of 24 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). Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges must be followed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. The leakage rate of each main steam isolation valve path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves). If both isolation valves in the penetration are closed the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with the Primary Containment Leakage Rate Testing Program). The combined leakage through all MSIV leakage paths must be ≤ 46 scfh when tested at ≥ 25 psig. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

MSIV leakage is considered part of L_a.

REFERENCES

1. Technical Requirements Manual.

2. UFSAR, Section 15.6.5.

3. UFSAR, Section 15.6.4.

4. UFSAR, Section 15.2.4.

5. UFSAR, Section 6.2.4.1.

6. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000.

Attachment C-1

MARKED-UP TECHNICAL SPECIFICATIONS AND BASES PAGES FOR PROPOSED CHANGE

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
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~~ A REPRESENTATIVE SAmple of reactor instrumentation line ~~EFCV~~ EFCVs actuate~~s~~ 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

A representative sample of

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valves (EFCVs) ~~is~~ Are OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break condition. This test is performed by blowing down the instrument line during an inservice leak or hydrostatic test and verifying a distinctive "click" when the poppet valve seats or a quick reduction in flow. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 24 month Frequency is based on the need to perform 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 experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

INSERT 1

INSERT 2

SR 3.6.1.3.9

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. The Frequency of 24 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). Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges must be followed.

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. The combined leakage rate for all MSIV leakage paths is $\leq$ 46 scfh when tested at $\geq$ 25 psig. The leakage rate of each main steam

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.10 (continued)

isolation valve path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves). If both isolation valves in the penetration are closed the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with the Primary Containment Leakage Rate Testing Program). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

MSIV leakage is considered part of L_a.

REFERENCES

1. Technical Requirements Manual.
2. UFSAR, Section 15.6.5.
3. UFSAR, Section 15.6.4.
4. UFSAR, Chapter 15.
5. UFSAR, Section 5.2.2.2.3.
6. UFSAR, Section 6.2.4.1.

INSERT 3

QUAD CITIES 1 AND 2

Insert 1:

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time.

Insert 2:

The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 7). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval 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.

Insert 3:

7. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000

Attachment C-2

TYPED PAGES FOR TECHNICAL SPECIFICATIONS CHANGE

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
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 a representative sample of reactor instrumentation line EFCVs actuate 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break condition. This test is performed by blowing down the instrument line during an inservice leak or hydrostatic test and verifying a distinctive "click" when the poppet valve seats or a quick reduction in flow. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 24 month Frequency is based on the need to perform 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 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 7). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval 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.

SR 3.6.1.3.9

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. The Frequency of 24 months

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

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). Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges must be followed.

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. The combined leakage rate for all MSIV leakage paths is ≤ 46 scfh when tested at ≥ 25 psig. The leakage rate of each main steam isolation valve path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves). If both isolation valves in the penetration are closed the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with the Primary Containment Leakage Rate Testing Program). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

MSIV leakage is considered part of L_a.

REFERENCES

1. Technical Requirements Manual.

2 UFSAR, Section 15.6.5.

3. UFSAR, Section 15.6.4.

4. UFSAR, Chapter 15.

5. UFSAR, Section 5.2.2.2.3.

6. UFSAR, Section 6.2.4.1.

7. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000.

Attachment D

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC, is requesting a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-19, DPR-25, DPR-29, and DPR-30, for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed change revises TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement 3.6.1.3.8 to allow a representative sample of reactor instrumentation line excess flow check valves (EFCVs) to be tested every 24 months, such that each EFCV will be tested nominally at least once every 10 years.

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

The current Technical Specification (TS) Surveillance Requirement (SR) frequency requires each reactor instrumentation line excess flow check valve (EFCV) to be tested every 24 months. The EFCVs at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) are designed to remain open during normal operation, but will close automatically in the event of an instrument line break downstream of the valve. The proposed change allows a reduced number of reactor instrumentation line EFCVs to be tested every 24 months. Industry operating experience demonstrates a high level of reliability for these EFCVs. A failure of an EFCV to isolate cannot initiate previously evaluated accidents (i.e., a break in a reactor coolant pressure boundary (RCPB) instrument line outside containment). Therefore, there is no increase in the probability of an accident as a result of this proposed change.

The postulated break of an instrument line connected to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Reports (UFSARs) for DNPS and QCNPS. 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 below the guidelines of 10 CFR 100, "Reactor Site Criteria." The NRC

approved General Electric Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," discusses through operating experience that there is a high degree of reliability with the EFCVs and that there are little radiological consequences resulting from an EFCV failure. The radiological consequences for an instrument line break do not credit the EFCVs for isolating the break. Therefore, the consequences of an instrument line break are not impacted by the proposed level of testing. Based on the above, the proposed TS change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change allows a reduced number of reactor instrumentation line EFCVs to be tested every 24 months. No other changes in requirements are being proposed. Industry operating experience as documented in NEDO-32977-A, provides supporting evidence that the reduced testing 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 testing is bounded by the evaluation of an instrument line break described in the UFSARs for DNPS and QCNPS. The proposed changes do not physically alter the plant and will not alter the operation of structures, systems, and components as described in the UFSARs. Therefore, a new or different kind of accident from any accident previously evaluated will not be created.

The proposed TS change does not involve a significant reduction in a margin of safety.

The consequences of an unisolable rupture of a RCPB instrument line outside containment has been previously evaluated in the UFSARs for DNPS and QCNPS. That evaluation assumed a continuous discharge of reactor coolant for the duration of the detection and cooldown sequence (i.e., no credit was assumed for isolating the break by the associated EFCV in the ruptured instrument line). Since a continuous discharge was assumed in this evaluation, any potential failure of the associated EFCV to isolate postulated by the reduced testing frequency is bounded. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Conclusion

Based upon the above evaluation, we have concluded that the three criteria of 10 CFR 50.92(c) are satisfied and that the proposed TS change involves no significant hazards consideration.

Attachment E

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC, is requesting a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-19, DPR-25, DPR-29, and DPR-30, for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. The proposed change revises TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.8 to allow a representative sample of reactor instrumentation line excess flow check valves (EFCVs) to be tested every 24 months, such that each EFCV will be tested nominally at least once every 10 years.

EGC has evaluated this proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that this proposed change meets the criteria for a categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a surveillance requirement, and the amendment meets the following specific criteria:

(i) The proposed change involves no significant hazards consideration.

As demonstrated in Attachment D, this proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change, which allows testing of a representative sample of reactor instrumentation line EFCVs every 24 months, is consistent with the plant design basis. There will be no significant increase in the amounts of any effluents released offsite. The proposed change does not result in an increase in power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed change will not affect the types or increase the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in changes in the configuration of the facility. The proposed change only affects the frequency of testing reactor instrumentation line EFCVs. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.