

**PECO NUCLEAR**

A Unit of PECO Energy

PECO Energy Company
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

May 31, 2000

Docket Nos. 50-277
50-278License Nos. DPR-44
DPR-56U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555Subject: Peach Bottom Atomic Power Station, Units 2 and 3
License Change Application ECR 00-00459

Dear Sir/Madam:

PECO Energy Company (PECO Energy) hereby submits License Change Application ECR 00-00459, in accordance with 10 CFR 50.90, requesting a change to the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Facility Operating Licenses. This proposed change will revise Surveillance Requirement (SR) 3.6.1.3.11 to require testing of "a representative" sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 10 years (nominally). Currently SR 3.6.1.3.11 requires testing of the EFCVs on a 24 month frequency.

Information supporting this request is contained in Attachment 1 to this letter, and the mark-up pages showing the proposed changes to the PBAPS, Units 2 and 3 TS are contained in Attachment 2. Attachment 3 contains the final pages.

A copy of this License Change Application, including the reasoned analysis about a no significant hazards consideration, is being provided to the appropriate Pennsylvania State official in accordance with the requirements of 10 CFR 50.91(b)(1).

We request that this amendment to the PBAPS, Units 2 and 3 TS be approved by August 25, 2000 for PBAPS, Units 2 and 3 in order to support the upcoming PBAPS, Unit 2 refueling outage.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

James A. Hutton
Director - Licensing

Enclosures: Affidavit, Attachment 1, Attachment 2, Attachment 3

cc: H. J. Miller, Administrator, Region I, USNRC
A. C. McMurtry, USNRC Senior Resident Inspector, PBAPS

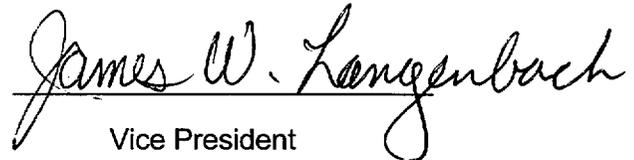
A001

R. R. Janati, Commonwealth of Pennsylvania

COMMONWEALTH OF PENNSYLVANIA :
 : SS.
COUNTY OF CHESTER :

J. W. Langenbach, being first duly sworn, deposes and says:

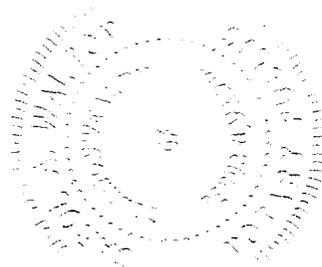
That he is Vice President of PECO Energy Company; the Applicant herein; that he has read the attached License Change Application ECR 00-00459, for Peach Bottom Facility Operating Licenses DPR-44 and DPR-56, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.


Vice President

Subscribed and sworn to
before me this 31st day
of May 2000.


Notary Public

Notarial Seal
Carol A. Walton, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires May 28, 2002
Member, Pennsylvania Association of Notaries



ATTACHMENT 1

**PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3**

**Docket Nos. 50-277
50-278**

**License Nos. DPR-44
DPR-56**

**LICENSE CHANGE APPLICATION
ECR 00-00459**

**Revision of Surveillance Requirement (SR) 3.6.1.3.11 Concerning Excess Flow Check Valves
(EFCVs)**

Supporting Information - 5 Pages

Introduction

PECO Energy Company, Licensee under Facility Operating Licenses DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended to revise Surveillance Requirement (SR) 3.6.1.3.11 to require testing of “a representative” sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 10 years (nominally). The TS pages showing the proposed changes are contained in Attachments 2 and 3. This License Change Application provides a discussion and description of the proposed changes, a safety assessment of the proposed changes, information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

Discussion and Description of the Proposed Changes

Surveillance Requirement (SR) 3.6.1.3.11 currently requires verification that each reactor instrumentation line Excess Flow Check Valve (EFCV) actuate to the isolation position on a simulated instrument line break signal every 24 months. PECO Energy Company proposes to revise Surveillance Requirement (SR) 3.6.1.3.11 to require testing of “a representative” sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 10 years (nominally). This proposed change is similar to existing performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50 Appendix J. The basis for this change is consistent with the Boiling Water Reactor (BWR) Owners Group Topical Report B21-00658-01, “Excess Flow Check Valve Testing Relaxation” by General Electric Nuclear Energy (GENE). The GENE Topical Report was approved by the NRC in a Safety Evaluation Report (SER) dated March 14, 2000. This proposed change is also consistent with TSTF-334 (“Relaxed Surveillance Frequency for Excess Flow Check Valve Testing”), which was also approved as part of the March 14, 2000 Safety Evaluation. The GENE report provides justification for a relaxation in the SR frequency. The report demonstrates, through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure.

PECO Energy is requesting approval of the proposed TS pages contained in Attachment 3 for both units.

Safety Assessment

The proposed changes will increase the test interval of the EFCVs. The GENE Topical Report compares this situation to Option B of Appendix J to 10 CFR 50. 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 the NRC SER dated March 14, 2000, the NRC accepted the test interval extension, which may be as great as 10 years, based on the EFCVs historically

high reliability, and their low risk significance, and radiological consequences should they fail.

EFCVs are used in the PBAPS, Units 2 and 3 reactor and primary system instrumentation lines to limit the release of fluid from the reactor coolant system in the event of an instrument line break. Examples of applications of EFCVs include reactor pressure vessel level/pressure instruments, and Reactor Core Isolation Cooling (RCIC) steam line flow instruments. Instrument lines connected to the Reactor Coolant Pressure Boundary (RCPB) are equipped with a ¼-inch flow-restricting orifice. The orifice size is selected by optimizing the minimum coolant release consistent with minimum effect on instrument response. A manually operated stop valve and an excess flow check valve are installed in each line outside containment as close as practicable to the primary containment penetration. As discussed in the PBAPS, Units 2 and 3 Safety Evaluation Report, dated August 11, 1972, the EFCVs provide isolation provisions consistent with the guidelines of the Supplement to Safety Guide 11, "Instrument Lines Penetrating Primary Reactor Containment Backfitting Considerations." The GENE topical report states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break outside of containment coincident with a design basis LOCA would be of sufficiently low probability to be outside of the design basis. In order to conform with the intent of the General Electric Topical Report associated with this change, the presence of restricting orifices will be confirmed before we reduce the amount of excess flow check valve testing on the particular line.

The GENE Topical Report provides detailed information about the results of EFCV surveillance testing at 12 BWR plants. These results represent a total of 12,424.5 valve operating years (1.09E+08 hours) with a plant average of 1035 valve years per plant. Considering the total number of EFCV failures (11) out of 1.09E+08 hours for the 12 BWRs, the Topical Report concluded that EFCVs had a low failure rate (1.01E-07/hour). In taking a similar representative time sample at PBAPS, Units 2 and 3 (938 valve years, or 8.22E+06 valve hours, for 68 and 66 valves at PBAPS, Units 2 and 3, (respectively) there were no EFCV surveillance test failures.

The EFCV manufacturer type installed for all but one of the valves at PBAPS, Units 2 and 3 is Dragon. Table 4-2 of the GENE Topical Report lists the best-estimate failure rate for the three plants using Dragon valves to be 9.2E-8/hour. This data is based on two recorded failures out of 21.8E+6 total operating hours at the three plants (Fermi 2: 8.15E+6 hours, Clinton: 1.93E+6 hours, WNP 2: 11.8E+6 Hours). At PBAPS, Units 2 and 3, a record search back to 1993 (8.22E+6 valve hours) did not reveal any surveillance test failures of EFCVs. The PBAPS, Units 2 and 3 data is consistent both in service time sampled, and reliability, with the results listed in the GENE report. Therefore, we have concluded that the GENE Report bounds the reliability of Peach Bottom's EFCVs.

At PBAPS, a feedback mechanism is in place to respond to EFCV failures which are identified during surveillance testing. The impact of EFCV failures on the system will be

monitored in accordance with the requirements established by the PBAPS Maintenance Rule Program. Degraded performance or Functional Failures of in-scope Structures, Systems or Components (SSCs) will be detected through existing performance criteria, which are established in accordance with the requirements of 10 CFR 50.65, NUMARC 93-01, and plant procedures. Once such degradation is detected, cause evaluations are performed to establish appropriate corrective actions.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Section 5.2.3.5. 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 10 CFR 100. 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.

As discussed in the methodology provided in the NRC SER to Duane Arnold Energy Center (Docket No. 50-331, dated December 29, 1999), the BWR Owners Group assumed a single instrument line break frequency of $3.52E-05/\text{year}$. This estimate was based on the EPRI Technical Report No. 100380, "Pipe Failures in U. S. Commercial Nuclear Power Plants," dated July 1992. This frequency corresponds to pipe sizes between $\frac{1}{2}$ inch to 2 inches in diameter and the BWR Owners Group considered these pipe sizes to represent the subject instrument line piping. Thus, for PBAPS, Units 2 and 3, the product of this single instrument line break frequency and the total number of instrument lines at PBAPS, Units 2 and 3, 68 and 66, respectively, resulted in a total plant instrument line break frequency estimate of $2.39E-3/\text{year}$ and $2.32E-3/\text{year}$, respectively. Using the surveillance interval for 2 years, the instrument line break frequency of $2.39E-3/\text{year}$ and $2.32E-3/\text{year}$, respectively, and the total plant EFCV failure frequency of $5.53E-3/\text{year}$ as provided in the NRC SER, the release frequency was estimated to be $1.32E-5/\text{year}$ and $1.28E-5/\text{year}$. For a surveillance interval of 10 years, the release frequency was estimated to be about $6.6E-5/\text{year}$ and $6.4E-5/\text{year}$, which depicts an increase of about $5.28E-5/\text{year}$ and $5.12E-5/\text{year}$ from the 2 year interval.

As compared to the release frequency provided in the NRC SER, this estimated increase in the release frequency is not significant. The NRC SER used estimated data from the Duane Arnold Energy Center (letter from K. E. Peveler (DAEC) to U. S. Nuclear Regulatory Commission, dated October 8, 1999) submittal. Duane Arnold's estimate was based on a random break of any of 94 instrument lines. PBAPS, Units 2 and 3 have a lesser number of applicable lines namely 68 and 66, respectively, therefore, the estimated increase for PBAPS, Units 2 or 3 is less than that already determined as "not significant" by the NRC.

Table 1 depicts the changes in release frequency estimates:

Table 1 Release Frequency Estimates

Unit	Estimate with 2 year surveillance tests	Estimate with 10 year surveillance tests	Change in release frequency estimate
2	1.32E-5/year	6.6E-5/year	+5.28E-5/year
3	1.28E-5/year	6.4E-5/year	+5.12E-5/year

The reduced testing associated with this proposed change will result in cost savings during outages, and dose savings during power operations and outages without significantly impacting the health and safety of the general public.

Information Supporting a Finding of No Significant Hazards Consideration

It is concluded that the proposed changes to the PBAPS, Units 2 and 3 TS do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed TS changes do 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 24 months. The EFCVs at PBAPS, Units 2 and 3 are designed to not close accidentally during normal operation, but will close automatically in the event of a line break downstream of the valve. The EFCVs are provided with position indication on a local panel. A general alarm is provided in the control room to indicate that an EFCV position has changed state. As discussed in the PBAPS, Units 2 and 3 Safety Evaluation Report, dated August 11, 1972, the EFCVs provide isolation provisions consistent with the guidelines of the Supplement to Safety Guide 11, "Instrument Lines Penetrating Primary Reactor Containment Backfitting Considerations." The BWROG Report B21-00658-01 demonstrates, through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. 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.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Section 5.2.3.5. 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 10 CFR 100. 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. Therefore, the proposed TS changes do not involve a

significant increase in the consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes allow 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 GENE 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 Section 5.2.3.5 of the UFSAR. The changes are 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 proposed TS changes do not involve a significant reduction in a margin of safety.

The consequences of an unisolable rupture of an instrument line has been previously evaluated in the PBAPS, Units 2 and 3 UFSAR, Section 5.2.3.5. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence. Since a continuous discharge was assumed in this evaluation, any potential failure of an EFCV to isolate postulated by this reduced testing frequency is bounded and does not involve a significant reduction in the margin of safety.

Information Supporting an Environmental Assessment

An environmental assessment is not required for the proposed changes since the proposed changes conform to the criteria for "actions eligible for categorical exclusion" as specified in 10 CFR 51.22(c)(9). The proposed changes will have no impact on the environment. The proposed changes do not involve a significant hazards consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

We have concluded that the proposed changes to the PBAPS, Units 2 and 3 TS do not involve a Significant Hazards Consideration.

ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

MARK-UP TECHNICAL SPECIFICATIONS CHANGES

Attached Pages

Units 2 and 3

3.6-15
Bases 3.6-28
Inserts

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	Verify the isolation time of each power operated and each automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.10	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.11	Verify each ^{a representative sample of} reactor instrumentation line EFCV actuates to the isolation position on a ^S simulated instrument line break signal.	24 months
SR 3.6.1.3.12	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.13	Verify the CAD System supplies nitrogen to the SGIG System upon loss of the normal air supply.	24 months

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.3.10 (continued)

FUNCTIONAL TEST in LCO 3.3.6.1 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 will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.11

a representative sample of

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 a simulated instrument line break signal. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. While this Surveillance can be performed with the reactor at power for some of the EFCVs, operating experience has shown that these components will 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 3.6.1.3.12

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 3.6.1.3.6).

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	Verify the isolation time of each power operated and each automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
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(continued)

BASES

SURVEILLANCE
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(continued)

Insert 1 to BASES for SR 3.6.1.3.11

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 sample 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 to BASES for SR 3.6.1.3.11

The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. 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.

For some EFCVs, this Surveillance can be performed with the reactor at power.

ATTACHMENT 3

**PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3**

**Docket Nos. 50-277
50-278**

**License Nos. DPR-44
DPR-56**

FINAL TECHNICAL SPECIFICATIONS CHANGES

Attached Pages

Units 2 and 3

**3.6-15
Bases 3.6-28**

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(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.6.1.3.10 (continued)

FUNCTIONAL TEST in LCO 3.3.6.1 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 will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) is OPERABLE by verifying that the valve actuates to the isolation position on a simulated instrument line break signal. 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 sample 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 so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. 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. For some EFCVs, this Surveillance can be performed with the reactor at power.

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