10 CFR 50.90

PECO NUCLEAR
A Unit of PECO Energy

PECO Energy Company
200 Exelon Way
Kennett Square, PA 19348

September 5, 2000

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Limerick Generating Station, Units 1 and 2
Technical Specifications Change Request 00-05-0

Dear Sir/Madam:

PECO Energy Company (PECO Energy) hereby submits Technical Specifications Change Request 00-05-0, in accordance with 10 CFR 50.90, requesting a change to the Limerick Generating Station (LGS), Units 1 and 2 Facility Operating Licenses. This proposed change will revise Surveillance Requirement 4.6.3.4 to require testing of a representative sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 120 months. Currently, Surveillance Requirement 4.6.3.4 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 LGS, Units 1 and 2 TS are contained in Attachment 2. Attachment 3 contains the final pages.

A copy of this Technical Specifications Change Request, 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).

A001

We request that this amendment to the LGS, Units 1 and 2 TS be approved by March 24, 2001 for LGS, Units 1 and 2 in order to support the upcoming LGS, 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. L. Burritt, USNRC Senior Resident Inspector, LGS
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 Technical Specifications Change Request 00-05-0 involving excess flow check valve testing for Limerick Generating Station Facility Operating Licenses NPF-39 and NPF-85, 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.

James W Langenbach

Vice President

Subscribed and sworn to

before me this 5th day

of September 2000.

Carol A. Walton

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

LIMERICK GENERATING STATION
UNITS 1 and 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
00-05-0

Revision of Surveillance Requirement 4.6.3.4 Concerning Excess Flow Check Valves (EFCVs)

Supporting Information - 5 Pages

September 5, 2000
LGS TSCR 00-05-0

Docket Nos. 50-352
50-353
License Nos. NPF-39
NPF-85

Introduction

PECO Energy Company, Licensee under Facility Operating Licenses NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2, requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended to revise Surveillance Requirement 4.6.3.4 to require testing of "a representative" sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 120 months. The TS pages showing the proposed changes are contained in Attachments 2 and 3. This Technical Specifications Change Request 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 4.6.3.4 currently requires verification that each reactor instrumentation line Excess Flow Check Valve (EFCV) shown in Table 3.6.3-1 be demonstrated OPERABLE at least once per 24 months by verifying that the valve checks flow. PECO Energy Company proposes to revise Surveillance Requirement 4.6.3.4 to require testing of "a representative" sample of Excess Flow Check Valves (EFCVs), shown in Table 3.6.3-1, such that each EFCV will be tested at least once every 120 months. This proposed change is similar to existing performance-based testing programs, such as inservice testing of snubbers, and Option B to 10 CFR 50 Appendix J. The basis for this change is consistent with the GE Nuclear Energy (GENE) Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation." 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 Surveillance Requirement 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 LGS, Units 1 and 2 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. Those instrument lines that do not connect to the RCPB conform to Regulatory Guide 1.11 in that they are either equipped with an EFCV or an isolation valve capable of remote operation from the control room, and are sized or orificed to meet the criteria outlined in Regulatory Guide 1.11. An excess flow check valve is installed in each line outside containment as close as practicable to the primary containment penetration. 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.

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 LGS, Units 1 and 2 (1808 valve years, or 1.57E+07 valve hours, for 113 valves per unit) there was only one EFCV surveillance test failure.

The EFCV's manufacturer type installed at LGS are either Dragon (72) or Marotta (154). 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-08/hour. Table 4-2 of the GENE Topical Report lists the best-estimate failure rate for the five plants using Marotta valves to be 8.37E-08/hour. At LGS, Units 1 and 2, a record search back to 1992 (1.57E+07 valve hours) did reveal one valve surveillance test failure of an EFCV. The LGS, Units 1 and 2 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 Limerick's EFCVs.

PECO Energy will revise our 10CFR50.65 Maintenance Rule Performance Criteria to ensure EFCV performance remains consistent with the extended test interval. The new performance criteria for each unit is less than or equal to 2 failures on a 24 month rolling average. The two year rolling average was chosen to be consistent with the existing LGS Maintenance Rule program and the 24 month refueling cycle. When the performance criteria is exceeded, a 10CFR50.65(a)(1) determination will be performed in accordance with station procedures.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Section 15.6.2. 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/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 ½ inch to 2 inches in diameter and the BWR Owners Group considered these pipe sizes to represent the subject instrument line piping. Thus, for LGS, Units 1 and 2, the product of this single instrument line break frequency and the total number of instrument lines at LGS, Units 1 and 2, 113 per unit, resulted in a total plant instrument line break frequency estimate of 3.98E-3/year per unit. Using the surveillance interval for 2 years, the instrument line break frequency of 3.98E-3/year per unit, and the total plant EFCV failure frequency of 5.53E-3/year as provided in the NRC SER, the release frequency was estimated to be 2.19E-5/year per unit. For a surveillance interval of 10 years, the release frequency was estimated to be about 1.10E-4/year per unit, which depicts an increase of about 8.78E-5/year from the 2 year interval. This increase is comparable to that accepted by the NRC in their safety evaluation dated March 14, 2000 for the GENE Report.

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
1	2.19 E-05/year	1.10 E-04/year	+8.78 E-05/year
2	2.19 E-05/year	1.10 E-04/year	+8.78 E-05/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 LGS, Units 1 and 2 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 Surveillance Requirement (SR) frequency requires each reactor instrumentation line Excess Flow Check Valve (EFCV) to be tested every 24 months. The EFCVs at LGS, Units 1 and 2 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 valve position indication in the reactor enclosure. A general alarm is provided in the control room to indicate that an EFCV position has changed state. As discussed in the LGS, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) (Section 6.2.4.3.1.5), instrument lines that penetrate the containment from the Reactor Coolant Pressure Boundary (RCPB) conform to Regulatory Guide 1.11 in that they are equipped with a restricting orifice located inside the drywell and an EFCV located outside the drywell as close as practical to the containment. The GE Nuclear Energy (GENE) Report 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 15.6.2. 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 Technical Specification (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 15.6.2 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 LGS, Units 1 and 2 UFSAR, Section 15.6.2. 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 LGS, Units 1 and 2 TS do not involve a Significant Hazards Consideration.

ATTACHMENT 2

LIMERICK GENERATING STATION
UNITS 1 and 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

MARK-UP TECHNICAL SPECIFICATIONS CHANGES

Attached Pages

Units 1 and 2

3/4 6-18
Bases B 3/4 6-4
Insert

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 24 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 ~~Each~~ A representative sample of reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 24 months, such that each valve is tested at least once every 120 months, by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 24 months by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested at least once per 120 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF INSERT

Vacuum relief valves are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. Two pairs of valves are required to protect containment structural integrity. There are four pairs of valves (three to provide minimum redundancy) so that operation may continue for up to 72 hours with no more than two pairs of vacuum breakers inoperable in the closed position.

Each vacuum breaker valve's position indication system is of great enough sensitivity to ensure that the maximum steam bypass leakage coefficient of

$$\frac{A}{\sqrt{k}} = 0.05 \text{ ft}^2$$

for the vacuum relief system (assuming one valve fully open) will not be exceeded.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 24 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 ~~Each~~ A representative sample of instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 24 months, such that each valve is tested at least once every 120 months, by verifying that the valve checks flow.

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DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

INSERT

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for the vacuum relief system (assuming one valve fully open) will not be exceeded.

INSERT

This Surveillance Requirement requires a demonstration that a representative sample of reactor instrument 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. 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 Surveillance Requirement 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 10 year interval is based on other performance-based testing programs. Furthermore, any EFCV failures will be evaluated to determine if additional testing in the 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

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

FINAL TECHNICAL SPECIFICATIONS CHANGES

Attached Pages

Units 1 and 2

3/4 6-18
Bases B 3/4 6-4
Bases B 3/4 6-4a

SURVEILLANCE REQUIREMENTS

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a. At least once per 31 days by verifying the continuity of the explosive charge.

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DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

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VACUUM RELIEF (Continued)

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for the vacuum relief system (assuming one valve fully open) will not be exceeded.

SURVEILLANCE REQUIREMENTS

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The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

This Surveillance Requirement requires a demonstration that a representative sample of reactor instrument 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. 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 Surveillance Requirement 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 10 year interval is based on other performance-based testing programs. Furthermore, any EFCV failures will be evaluated to determine if additional testing in the 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.

3/4.6.4 VACUUM RELIEF

Vacuum relief valves are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

VACUUM RELIEF (Continued)

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. Two pairs of valves are required to protect containment structural integrity. There are four pairs of valves (three to provide minimum redundancy) so that operation may continue for up to 72 hours with no more than two pairs of vacuum breakers inoperable in the closed position.

Each vacuum breaker valve's position indication system is of great enough sensitivity to ensure that the maximum steam bypass leakage coefficient of

$$\frac{A}{\sqrt{k}} = 0.05 \text{ ft}^2$$

for the vacuum relief system (assuming one valve fully open) will not be exceeded.