March 8, 2002

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: ONE-TIME DEFERRAL OF CONTAINMENT INTEGRATED LEAK RATE TEST AND DRYWELL-TO-SUPPRESSION CHAMBER BYPASS LEAKAGE TEST (TAC NOS. MB2894 AND MB2895)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 202 to Facility Operating License No. NPF-14 and Amendment No. 176 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 30, 2001, as supplemented by letters dated September 7, October 16, and December 5, 2001, and January 18, 2002.

These amendments revise TSs to allow a one-time deferral of the Type A containment integrated leak rate test (ILRT) and a one-time deferral of the drywell-to-suppression chamber bypass leakage test. The Unit 1 ILRT may be deferred to no later than May 3, 2007, and the Unit 2 ILRT may be deferred to no later than October 30, 2007, resulting in an extended interval of up to 15 years for performance of these tests at each unit. The drywell-to-suppression chamber bypass leakage test will continue to be conducted with the ILRT at each unit; consistent with current practice.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

***/RA/***

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 202 to License No. NPF-14
2. Amendment No. 176 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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Units 1 & 2

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PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 202
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

    A. The application for the amendment filed by PPL Susquehanna, LLC, dated July 30, 2001, as supplemented by letters dated September 7, October 16, and December 5, 2001, and January 18, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

    C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 8, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 202

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the PPL Susquehanna, LLC, dated July 30, 2001, as supplemented by letters dated September 7, October 16, and December 5, 2001, and January 18, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 176 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 8, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. NPF-22

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

## 1.0 INTRODUCTION

By letter dated July 30, 2001, as supplemented by letters dated September 7, October 16, and December 5, 2001, and January 18, 2002 (collectively, the application; References 1 - 5, respectively), PPL Susquehanna, LLC (the licensee), submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2 (SSES-1 and 2), Technical Specifications (TSs). The requested changes would revise TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time deferral of the Type A containment integrated leakage rate test (Type A, or ILRT) at SSES-1 and 2. The Unit 1 test would be deferred to no later than May 3, 2007, and the Unit 2 test would be deferred to no later than October 30, 2007, resulting in an extended interval of 15 years for performance of the next ILRT at each unit. Additionally, the proposed amendments would allow a one-time deferral of the drywell-to-suppression chamber bypass leakage test, Surveillance Requirement (SR) 3.6.1.1.2, so that it would continue to be conducted along with the ILRT, consistent with current practice. The current TSs tie the interval of SR 3.6.1.1.2 to the interval of the ILRT. This amendment request has no effect on the frequency of the separate tests of the vacuum breaker valves alone (SR 3.6.1.1.3). That testing will continue to be performed every refueling outage, except when the full drywell-to-suppression chamber bypass leakage test of SR 3.6.1.1.2 is performed.

The TS change was requested based on the risk-informed approach developed using Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” This safety evaluation (SE) addresses the potential impact of the requested surveillance interval extensions on the licensee’s existing programs to inspect and monitor aging degradation of the containment pressure boundary.

## 2.0 BACKGROUND

SSES-1 and 2 utilize a General Electric boiling water reactor (BWR) Mark II primary containment structure. Each containment consists of a steel-lined, reinforced-concrete drywell and pressure suppression chamber (PSC) and an integrally-reinforced, concrete slab separating the drywell from the PSC. There are a number of downcomer vent pipes penetrating the slab and extending into the water of the PSC that are used for venting the

drywell atmosphere following a loss-of-coolant accident (LOCA). The containment is penetrated by access penetrations, and other process piping and electrical penetrations.

The integrity of the penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, and the overall leak-tight integrity of the containment is verified through an ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident pressure. The last ILRTs for SSES-1 and 2 were performed in May and October of 1992, respectively. With the requested surveillance interval extensions for both units, the next ILRTs and drywell-to-suppression chamber bypass leakage tests for SSES-1 and 2 will be performed no later than May and October of 2007, respectively.

## 2.1 Type A Test Interval Extension

Appendix J, Option B of 10 CFR Part 50, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. SSES-1 and 2, TS 5.5.12, requires that a program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. TS 5.5.12 further requires that this program shall be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by exceptions set forth in the TSs. RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute's (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995 (NEI 94-01).

For the Type A test (also known as an ILRT) of the containment structure, NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at SSES-1 and 2 have been successful, so their current interval requirement is 10 years.

The licensee is requesting additions to TS 5.5.12, "Primary Containment Leakage Rate Testing Program," which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS for Unit 1 says that the first Type A test performed after the May 4, 1992, Type A test shall be performed no later than May 3, 2007. The proposed TS for Unit 2 says that the first Type A test performed after the October 31, 1992, Type A test shall be performed no later than October 30, 2007.

## 2.2 Drywell-to-Suppression Chamber Bypass Leakage Test Interval

The Nuclear Regulatory Commission (NRC) issued License Amendment Nos. 160 and 131 to the operating licenses for SSES-1 and 2, respectively, on September 6, 1996 (Reference 6). These amendments changed the TSs such that the drywell-to-suppression chamber bypass leakage test frequency is tied to the Appendix J, Type A test frequency. Previously, the drywell-to-suppression chamber bypass leakage test interval had been 40 ± 10 months. Thus, the licensee's current request for a one-time, Type A test interval extension to 15 years also extends the drywell-to-suppression chamber bypass leakage test interval to 15 years.

The NRC staff's SE for License Amendment Nos. 160 and 131 assumed that the maximum Type A test interval would be 10 years (plus 15 months, potentially). To summarize that SE, the staff focused on (a) the licensee's complete identification and analysis of potential leakage paths to the suppression chamber; (b) historical bypass leakage test results; and, (c) the ability to mitigate a suppression chamber steam bypass event should one occur.

Potential Leakage Pathways

During a small break LOCA, potential leak paths between the drywell and suppression chamber airspace could result in excessive containment pressure, since the steam flow into the airspace would bypass the vapor suppression capabilities of the pool. The potential leakage paths between the drywell and the suppression chamber are: 1) piping which passes through the suppression chamber; 2) the diaphragm slab and the seal between it and the suppression chamber; 3) downcomer penetrations; 4) safety/relief valve discharge line penetrations; and 5) drywell-to-suppression chamber vacuum breakers. The NRC staff found that the drywell-to-suppression chamber vacuum breakers were by far the most likely potential leakage paths, and that the other four paths were unlikely to leak. Further, the drywell-to-suppression chamber vacuum breakers are subjected to LLRTs during each refueling outage in which the drywell-to-suppression chamber bypass leakage test is not performed.

Historical Bypass Leakage Test Results

Regarding testing history, the NRC staff found that past test results indicated that bypass leakage through passive potential leakage pathways (i.e. non-vacuum breaker pathways) had consistently been small, and thus provided a reasonable basis to conclude that these pathways constituted a minor source of bypass leakage.

Ability to Mitigate a Suppression Chamber Steam Bypass Event

The NRC staff found that the containment pressure response to the postulated bypass leakage could be mitigated by manually actuating the containment sprays. First the operators would activate the suppression chamber sprays, and then, if necessary, the drywell sprays.

The NRC staff concluded that the proposal to change the drywell-to-suppression chamber bypass leakage test frequency from once every 40 ± 10 months to a frequency in accordance with Appendix J, Option B, was acceptable. This was based on the following facts: (a) that bypass leakage through passive components had historically been much lower than the TS limit; (b) that such components/penetrations had a relatively low potential for leakage; and, (c) that the drywell-to-suppression chamber vacuum breakers would be tested on a frequency (every refueling outage) sufficient to identify and correct any excessive leakage through these components in a timely manner.

## 3.0 EVALUATION

### 3.1 Type A Test Interval Extension

#### 3.1.1 Inspection and Testing Program

The ILRT, the LLRTs, and inservice inspection (ISI) of the containment collectively ensure the leak-tight and structural integrity of the containment. In order to fully evaluate the requested TS changes, the NRC staff requested additional information regarding the licensee's program for containment ISI. The NRC staff also requested that the licensee address potential areas of weakness in its containment that might not be apparent in the risk assessment. The following is a discussion of the licensee's responses to the NRC staff's questions (References 3 and 4) regarding the proposed TS changes.

The licensee is using the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of Section XI of the ASME *Boiler and Pressure Vessel Code* (ASME Code), with approved relief from certain Code requirements, for conducting the ISI of the SSES-1 and 2 containments. The start dates for the current containment ISI 10-year intervals were March 2000 (Unit 1), and March 1999 (Unit 2). The application states that the accessible areas of the containment pressure boundary will be periodically monitored for signs of degradation. Additionally, the NRC staff notes that visual examinations are required by the existing SR 3.6.1.1.1, which is unchanged by the proposed amendments.

In response to an NRC staff question on the examination of the primary containment pressure boundary seals, gaskets, and bolts (Type B penetrations), the licensee stated that the initial frequency of Type B tests for penetrations and their components is 30 months. If the leakage rates from two consecutive as-found Type B tests are less than their established administrative limits, the test interval is extended to 60 months. If the leakage rates from three consecutive as-found tests are found to be less than the administrative limit for the penetration, then the test interval is extended to 120 months. In the event that a Type B penetration leakage rate is higher than the administrative limit, the test interval is reestablished at 30 months. Moreover, the licensee stated that, regardless of the above schedule, it performs a post-maintenance Appendix J, Type B test following any repair or disassembly of a component with a seal, gasket, or bolted connection. The NRC staff notes that such tests following repair, modification, or replacement of containment components are required by Subarticle IWE-5000 of the ASME Code, Section XI. Thus, an ILRT might be required to confirm that these activities are adequate and that further degradation does not exist in other areas of the containment. Additionally, the licensee is required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73.

In a follow-up question, the NRC staff asked the licensee to discuss other separately scheduled ISIs performed on seals, gaskets, and bolts in Type B penetrations (Reference 4). The licensee stated that separate ISIs are not conducted for these components. However, the administrative leakage rate limit for Type B testing (except for airlocks) is 0.5 standard liters per minute (SLM), which is very small, and the trend of Type B testing indicates that even minor degradation of these components has been identified through Type B testing. On these bases, the NRC staff finds that the licensee has established adequate procedures to identify degradation of these components.

In response to an NRC staff question related to the effects of degradation in uninspectable areas of the containment liners (i.e., those areas where visual inspection cannot be performed), the licensee considered the consequences of such an occurrence in a risk impact assessment of extending the Type A test interval to 15 years. In performing its risk analysis, the licensee made a number of assumptions (Reference 5):

(1) The containment leakage rate due to concealed corrosion is assumed to be a large leakage rate and analyzed as a Class 7 accident (Reference 7).
(2) The Type A ILRT will fail due to through-liner corrosion under 62 pounds per square inch absolute (psia) pressure.
(3) The leakage rate of the failed containment will be 100 La (i.e., 100 times the acceptable leakage rate during ILRT).
(4) The failure rate is calculated on the basis of two plant events in 70 reinforced concrete containments where through-liner corrosion has been identified.

The NRC staff finds the above assumptions used by the licensee in its analysis to be conservative. The following section discussed the licensee's risk assessment in further detail.

3.1.2 Risk Impact Assessment of Extending the Type A Test Interval to 15 Years

The risk impact assessment was provided to the NRC staff in the July 30, 2001, application (Reference 1). Additional analysis was provided in the December 5, 2001, and January 18, 2002, supplemental submittals (References 4 and 5). In performing the risk assessment, they considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute's (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," (Reference 7), and RG 1.174.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time from 18 to 60 months that a leak, detectable only by a Type A test, goes undetected. Since Type A tests only detect about 3 percent of leaks (the rest are identified during LLRTs based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage, in percent of person-rem/year, for the pressurized water reactor and BWR representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk ranging from 0.02 to 0.14 percent.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Appendix J, Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Appendix J, Option B rulemaking.

RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. This RG defines very small changes in the risk-acceptance guidelines as (a) increases in core damage frequency (CDF) less than $10^{-6}$ per reactor year, and (b) increases in large early release frequency (LERF) less than $10^{-7}$ per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3-in-10-year test interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee has provided information for estimating the change in the conditional containment failure probability to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology is consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. The increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

   A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.02 percent. The increase in the total integrated plant risk, given the change from a 3-in-10-year test interval to a 1-in-15-year test interval, was found to be 0.05 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3-in-10-year test interval to a 1-in-10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.

   The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is estimated to be $4 \times 10^{-10}$/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be $1 \times 10^{-9}$/year. Thus, in accordance with the RG 1.174 guidance, increasing the Type A interval to 15 years is considered to be a very small change in LERF.

3. The defense-in-depth philosophy is maintained.

   RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to increase by 0.0021 for the proposed change and 0.0063 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC finds that the defense-in-depth philosophy is maintained based on the very small change in the conditional containment failure probability for the proposed change.

   The NRC staff recognizes the limitations of a conditional containment failure probability approach. For plants such as SSES-1 and 2, with core damage frequency estimates well below $10^{-4}$, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. The licensee has sufficiently demonstrated that the total LERF is less than $10^{-5}$ for the purpose of this evaluation.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidance while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.2 Drywell-to-Suppression Chamber Bypass Leakage Test Interval

The conclusions reached by the NRC staff in the SE supporting License Amendment Nos. 160 and 131 (see section 2.2 above) remain valid. Further, the licensee has collected additional testing results since 1996 and has made the following statements.

The calculated acceptable total vacuum breaker leakage is 1,580,000 standard cubic centimeters per minute (sccm) or 55.8 standard cubic feet per minute (scfm). The highest total vacuum breaker leakage measured to date is 97,782 sccm, or 6.2% of the allowable leakage.

The calculated allowable leakage flow rate for an individual vacuum breaker set (in one line) is 632,000 sccm or 22.3 scfm. The largest leakage flow rate measured to date for an individual vacuum breaker set is 85,726 sccm, or 13.6% of the allowable leakage.

No drywell vacuum breakers have failed the leakage test or required corrective action due to leakage since the testing was started in 1993. These valves will continue to be locally tested at every refueling outage, except when the full drywell-to-suppression chamber bypass leakage test is performed.

To support a test interval increase to 15 years from the previously approved 10 years for the full drywell-to-suppression chamber bypass leakage test, the licensee has performed a risk-impact assessment. The assessment was provided to the NRC staff in the December 5, 2001, supplemental submittal (Reference 4). In performing the risk assessment, they considered the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, (Reference 7) the results and findings of the Susquehanna Individual Plant Examination, and RG 1.174.

The following steps were used to complete the risk assessment:

(a) Estimate the plant risk based upon the original 3-year bypass test interval;

Determine sequences that require bypass area in accordance with the plant's design basis. The bypass area analysis is performed in order to maintain the pressure suppression function during a small break LOCA. Loss of pressure suppression function results in containment overpressure failure prior to core damage;

(b) Calculate the probability of failure of bypass area;

(c) Calculate the probability of operator failure to use containment sprays;

(d) Calculate risk for 10-year bypass interval;

(e) Calculate risk for 15-year bypass interval;

(f) Calculate change in pertinent risk metrics.

The licensee provided an analysis which allowed for the estimation of the following risk metrics. The following conclusions can be drawn from the analysis associated with extending the drywell-to-suppression chamber test frequency:

1. The increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

   A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.04 percent. The increase in the total integrated plant risk, given the change from a 3-in-10-year test interval to a 1-in-15-year test interval, was found to be 0.11 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3-in-10-year test interval to a 1-in-10-year interval for Type A tests. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change in the drywell-to-suppression chamber bypass leakage test interval is considered small and supportive of the proposed change.

2. Increasing the drywell-to-suppression pool test interval to 15 years is considered to be a very small change in LERF.

   Since the drywell-to-suppression pool test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the drywell-to-suppression pool test interval from 1 in 10 years to 1 in 15 years is estimated to be $7 \times 10^{-11}$/year. The increase in LERF resulting from a change in the drywell-to-suppression pool test interval from the original 3 in 10 years to 1 in 15 years is estimated to be $2 \times 10^{-10}$/year. Thus, in accordance with the RG 1.174 guidance, increasing the drywell-to-suppression pool test interval to 15 years is considered to be a very small change in LERF.

3. The defense-in-depth philosophy is maintained.

   RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to increase by 0.00019 for the proposed change and 0.00056 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the very small change in the conditional containment failure probability for the proposed change.

   The NRC staff recognizes the limitations of a conditional containment failure probability approach. For plants, such as SSES-1 and 2, with core damage frequency estimates well below $10^{-4}$, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. The licensee has sufficiently demonstrated that the estimated total LERF supports the proposed change.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidance while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

### 3.3 Summary

Based on the licensee's procedures to preclude excessive degradation of the primary containment components discussed above, and incorporation of certain degradation in the risk analysis, the NRC staff finds that granting the requested ILRT extension will not adversely affect the leak tight integrity of the primary containment. The NRC staff finds that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradation of the pressure-retaining components of the SSES-1 and 2 primary containments. Therefore, the NRC staff finds that the interval until the next Type A test and the next drywell-to-suppression chamber bypass leakage test (SR 3.6.1.1.2) at SSES-1 and 2 may be extended to 15 years, and that the proposed TS changes are acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no

public comment on such finding (67 FR 5330). Accordingly, the amendments meet eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

## 7.0 REFERENCES

1. Letter from G. T. Jones to NRC, “Susquehanna Steam Electric Station Proposed Amendment No. 241 to License NFP[NPF]-14 and Proposed Amendment No. 206 to License NFP[NPF]-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) PLA-5342,” July 30, 2001 (ADAMS Accession No. ML012150121)

2. Letter from R. G. Byram to NRC, “Susquehanna Steam Electric Station Supplement to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) PLA-5361,” September 7, 2001 (ADAMS Accession No. ML012570463)

3. Letter from R. G. Byram to NRC, “Susquehanna Steam Electric Station Supplement No. 2 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) PLA-5380,” October 16, 2001 (ADAMS Accession No. ML012950364)

4. Letter from R. G. Byram to NRC, “Susquehanna Steam Electric Station Supplement No. 3 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) and the Drywell-to-Suppression Chamber Bypass Leakage Test SR 3.6.1.1.2 PLA-5408,” December 05, 2001 (ADAMS Accession No. ML013540269)

5. Letter from R. G. Byram to NRC, “Susquehanna Steam Electric Station Supplement No. 4 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) and the Drywell-to-Suppression Chamber Bypass Leakage Test SR 3.6.1.1.2 PLA-5424,” January 18, 2002 (ADAMS Accession No. ML020230142)

6. The Nuclear Regulatory Commission (NRC) issued License Amendment Nos. 160 and 131 to the operating licenses for SSES-1 and 2, respectively, on September 6, 1996

(Accession No. ML010110174)

7. EPRI TR-104285, “Risk Assessment of Revised Containment Leak Rate Testing Interval,” August 1994

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Date: March 8, 2002

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

***/RA/***

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures: 1. Amendment No. 202 to License No. NPF-14
2. Amendment No. 176 to License No. NPF-22
3. Safety Evaluation

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