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TALEN ENERGY

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

10 CFR 50.90

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON TECHNICAL SPECIFICATION CHANGES TO ADOPT TRAVELER TSTF-425
PLA-7334

Docket Nos. 50-387 and 50-388

References:

1. *Letter PLA-7119, [Proposed Amendments to License NPF-14 and NPF-22] Adoption of Technical Specification Task Force Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Task Force (RITSTF) Initiative 5," dated October 27, 2014 (Accession ML14317A052).*

2. *NRC Letter, "Request for Additional Information re: License Amendment Request to Adopt Technical Specifications Task Force Traveler (TSTF)-425, (TAC Nos. MF5151 and MF5152)," dated May 22, 2015 (Accession ML15103A396).*

The purpose of this letter is for Susquehanna Nuclear, LLC to provide the requested additional information (RAI). By Reference 1, Susquehanna Nuclear, LLC submitted a license amendment request (LAR) to modify Susquehanna Steam Electric Station, Units 1 and 2 (SSES) Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program. The program will implement Nuclear Energy Institute (NEI) 04-10, *"Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies,"* (Accession ML071360456). The changes adopt an NRC approved Technical Specification Task Force (TSTF) traveler, TSTF-425, Revision 3, (Accession ML080280275).

In Reference 2, the RAI includes seven questions for which responses are provided in Attachment 1. The revised, marked-up TS pages are included in Attachment 2. Revised, marked-up TS Bases pages are provided for information in Attachment 3.

Susquehanna Nuclear, LLC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in Reference 1. The additional information provided by this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. Furthermore, the additional information also does not affect the bases for concluding that neither an environmental impact statement nor an

environmental assessment needs to be prepared in connection with the proposed amendment.

There are no new regulatory commitments associated with this response.

If you have any questions or require additional information, please contact Mr. Jeffery N. Grisewood (570) 542-1330.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 2, 2015

Sincerely,

J. A. Franke

Attachments:
1. Response to Requested Additional Information
2. Revised Marked-up TS Pages
3. Revised Marked-up TS Bases Pages (For Information)

Copy: NRC Region I
Mr. J. E. Greives, NRC Sr. Resident Inspector
Mr. J. A. Whited, NRC Project Manager
Mr. B. Fuller, PA DEP/BRP

Attachment 1 to PLA-7334

Response to Requested Additional Information

Response to Requested Additional Information

By letter dated October 27, 2014 [(1)] Susquehanna Nuclear, LLC submitted a license amendment request (LAR) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The NRC requested additional information (RAI) in a letter dated May 22, 2015.[(2)] This Attachment provides the requested additional information.

RAI 1:

Finding and Observation (F&O) 1-12, for Supporting Requirement (SR) SC-A5, located on page 15 of 69 of Attachment 2 of the licensee's application, was developed because the licensee did not perform an evaluation to determine if certain accident sequences should be extended beyond 24 hours. The licensee addressed this F&O by adding section 2.1.17 to the success criteria notebook to outline the dominant considerations contributing to the 24 hour probabilistic risk assessment (PRA) mission time and the systems/equipment with less than 24 hour mission time. However, the licensee did not explain whether one of the three methods proposed by the peer review team was used. Furthermore, the ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Standard requires the licensee to use a minimum mission time of 24 hours for sequences in which safe and stable plant conditions have been achieved. The ASME/ANS Standard also states that the licensee should perform additional evaluation or modeling using an appropriate technique (the standard lists some examples), for sequences in which stable plant conditions would not be achieved by 24 hours.

a. Explain whether a safe and stable plant condition would be achieved for the accident sequences with the assumed 24 hour PRA mission time.

b. If not, explain the treatment of those accident sequences for which stable plant conditions would not be achieved within 24 hours, consistent with the ASME/ANS Standard guidelines.

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(1) Letter (PLA-7119), [Proposed Amendments to License NPF-14 and NPF-22] Adoption of Technical Specification Task Force Traveler TSTF-425, Revision 3, *"Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Task Force (RITSTF) Initiative 5,"* dated October 27, 2014 (Accession ML14317A052).

(2) NRC Letter, *"Request for Additional Information re: License Amendment Request to Adopt Technical Specifications Task Force Traveler (TSTF)-425, (TAC Nos. MF5151 and MF5152),"* dated May 22, 2015 (Accession ML15103A396).

SSES Response to RAI 1:

The Susquehanna PRA model is developed such that a safe and stable plant condition is achieved for modeled accident sequences with the assumed 24 hour PRA mission time. However, modeling of certain accident sequences is extended beyond 24 hours when required to reach safe and stable plant conditions.

F&O 1-12 discusses potential evaluation areas where modeling of certain accident sequences is extended beyond 24 hours when required to reach safe and stable plant conditions. Examples of injection from the condensate storage tank (CST) / refueling water storage tank (RWST), loss of decay heat removal (DHR), and room cooling success criteria were provided. Each of these examples with respect to the treatment in the Susquehanna PRA model is discussed in turn.

The event tree and success criteria notebook includes consideration of equipment run time and potential operator actions ensuring system operation success. Event sequences affected by the depletion of finite inventory injection sources such as that provided by the RWST and CST are included in the applicable event tree sequence review. For example, in station black-out (SBO) scenarios with other actions successful to maintain reactor core isolation cooling system (RCIC) availability, RCIC is made unavailable at about 15 hours in scenarios without CST refill from the RWST. Those same SBO scenarios with no DHR but with CST refill can support RCIC injection for about 30 hours. Assurance of a safe stable state (including the time after RCIC injection is lost at 15 hours or at 30 hours) is required to reach an OK end state in both cases.

Similarly, in other loss of DHR scenarios, containment venting and continued injection are required to reach a safe stable state. If containment venting fails, then additional consideration is included in the event sequence modeling regarding the impacts of long term containment failure (which is predicted to occur > 24 hours) on the available injection systems to ensure a safe stable state is reached.

The event tree and success criteria notebook was revised to include specific documentation regarding Susquehanna's treatment and assumptions supporting the 24 hour mission time, and for systems with less than a 24 hour mission time. This additional discussion validates the choice of a 24-hour mission time as being appropriate for other aspects of the PRA model applicable to room cooling dependencies and the "run-time" used for the fail to run events consistent with the ASME/ANS Standard guidelines.

RAI 2:

F&O 7-1, related to SR HR-C3, located on page 19 of 69 of Attachment 2 of the licensee's application, was developed because the licensee did not include an analysis of miscalibration for the component failure rate data. The licensee stated that adding this level of detail would have a small effect on the outcome of the PRA per NUREG-1792, *"Good Practices for Implementing Human Reliability Analysis."*[3] However, miscalibration of standby systems is an important consideration for surveillance test interval changes. Explain how miscalibration is modeled in the PRA in accordance with SR HR-C3.

SSES Response to RAI 2:

The text of SR HR-C3 from ASME/ANS RA-Sa-2009[4] reads:

> *"INCLUDE the impact of miscalibration as a mode of failure of standby systems."*

Miscalibration as a mode of failure of initiation of standby systems is addressed by the Susquehanna pre-initiator process and this type of event is considered for evaluation and inclusion in the PRA model. The pre-initiator methodology identifies components susceptible to common mode errors. At least one event will always be retained for any component / function that is susceptible to common mode failures. Risk significant events are evaluated further using a detailed quantification while non-risk significant events are retained in the model with their scoping values.

As indicated in the resolution to F&O 7-1 in Attachment 2 of the submittal,[5] the discussion from NUREG-1792 states *"Generally, unless the failure can effect multiple items, either missing the failure or double-counting the failure have small effects on the outcome of the PRA."* It is noted that the methodology is consistent with this philosophy as it does include miscalibration events for components whose failure could impact multiple items, but adding an increased level of detail for single component miscalibration events is not warranted as it would not significantly impact the risk assessment for any surveillance test interval change. The HRA documentation clarifies that these single miscalibration events are not included because they are very low contributors.

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(3) NUREG-1792, *"Good Practices for Implementing Human Reliability Analysis,"* dated April 30, 2005 (Accession ML051160213).

(4) American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-Sa-2009), Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, February 2009.

(5) Reference 1, Attachment 2 of letter PLA-7119, *"Documentation of PRA Technical Adequacy,"* (Accession ML14316A606)

RAI-3:

F&O 7-4, located on page 18 of 69 of Attachment 2 of the licensee's application, related to SR HR-B2, cites screening of common mode failures due to staggered testing/maintenance principles. Please clarify the resolution of this F&O.

SSES Response to RAI 3:

The text of SR HR-B2 from ASME/ANS RA-Sa-2009 reads:

> *"DO NOT screen activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems (HR-A3)."*

The text from the referenced SR HR-A3 reads:

> *"IDENTIFY the work practices identified above (HR-A1, HR-A2) that involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems [e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system (e.g., SLCS)]."*

The issue identified in the F&O was concerned with the use of staggered testing / maintenance principles for screening purposes. The intent of the guidance in the applicable SRs is to identify single maintenance acts that realign a valve or valves that would disable the entire system (i.e., the redundant divisions / trains) simultaneously, not consecutively. This is supported by the examples provided in the SR, specifically the example event related to the "realignment of an entire system", such as standby liquid control system (SLCS). When this interpretation of the SR is applied, the pre-initiator identification process would preclude the development of common mode misalignment events for like components in separate divisions. Because of this, it was not necessary to rely on staggered testing / maintenance principles for screening purposes. The intent of the F&O resolution was to indicate that there is no longer any reliance on staggered testing / maintenance principles for screening purposes.

RAI-4:

The licensee stated that the internal fires risk and external events assessment is based on the Individual Plant Examination of External Events (IPEEE) studies, but did not explain if the IPEEE reflects current plant configuration. Explain whether the licensee's evaluation of fire risk and other external events supporting this application reflects the current plant configuration and operating experience.

SSES Response to RAI 4:

As indicated in Section 3.1 of Attachment 2 in the submittal(5), it is recognized that the use of the available fire risk information from the IPEEE is limited, but the NEI 04-10 methodology allows a qualitative screening or bounding analysis to provide justification for acceptability of proposed surveillance frequency changes. Therefore, the intent is not to directly use any numerical results from the IPEEE fire studies or other external events, but to qualitatively assess any available information to determine the impact on proposed surveillance interval changes consistent with the NEI 04-10(6) methodology. The qualitative information and/or the bounding analysis from the internal events analysis as described in Section 3.5 of the submittal must be acceptable and reflective of the current plant configuration and operating experience. This information is documented for each surveillance test interval (STI) change provided to the Integrated Decision Panel. Therefore, if the qualitative assessment is used it will be acceptable and reflective of the current plant configuration and operating experience.

RAI-5:

Regulatory Guide 1.177, *"An Approach for Plan-Specific, Risk-Informed Decisionmaking: Technical Specification,"*(7) requires the licensee to provide additional details in terms of separating the failure rate contributions into demand-related and standby time-related contributions. It is not clear whether these terms were separated. Explain how the licensee intends to model failure rate contributions to surveillance frequencies using NEI 04-10 guidance.

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(6) Nuclear Energy Institute (NEI), *"Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies,"* Industry Guidance Document, NEI 04-10, Revision 1, April 2007 (Accession ML071360456).

(7) Regulatory Guide (RG) 1.177, Revision 1, *"An Approach for Plant-Specific, Risk-Informed Decisionmaking Technical Specifications,"* dated May 31, 2011, (Accession ML100910008).

SSES Response to RAI 5:

The failure rate contributions for the STI changes will follow the guidance delineated in Step 8 of NEI 04-10. That is, the standby time related failures will be assessed by direct change in the test interval for those SSCs that include a standby periodically tested failure mode in the Susquehanna PRA models along with the appropriate adjustments to common cause failure events. Where there is no standby periodically tested event in the PRA models and one is not added or the failure cannot be divided into time based and non-time based contributions, as allowed by RG 1.177, all contributors to the failure rate will be assumed to be time based and the independent and common cause values adjusted accordingly.

If the SSCs do not appear explicitly in the PRA models, then either a bounding assessment using a surrogate event or a qualitative assessment will be performed in accordance with the NEI 04-10 guidance.

RAI-6:

As part of the license amendment request (LAR), the licensee proposed to remove the definition of "STAGGERED TEST BASIS" from TS Section 1.1, "Definitions," and relocate this frequency from the TS to the Surveillance Frequency Control Program. However, the term "STAGGERED TEST BASIS" is retained in TS 5.5.14, "Control Room Envelope Habitability Program," specifically in TS 5.5.14.d. Please provide a justification for the above stated discrepancy.

SSES Response to RAI 6:

Because the Control Room Envelope Habitability Program will continue to use the term "STAGGERED TEST BASIS," it does not need to be removed from TS Section 1.1, "Definitions." Revised marked up TS pages for Units 1 and 2 (e.g., page 1.1-6) are provided to show this definition will not be removed/changed in the TS. This is an administrative deviation from TSTF-425 with no impact on the NRC's model safety evaluation.

RAI-7:

During review of the proposed changes to Surveillance Requirement (SR) 3.6.1.1.2 the NRC staff notes that the licensee's proposed change is not consistent with TSTF-425. Specifically, the licensee's proposal would relocate the requirement to perform the surveillance when the condition in the note is met to the Surveillance Frequency Control

Program. However, TSTF-425 retains that frequency in the TS. Please provide further justification for this change.

SSES Response to RAI 7:

No change to the existing surveillance frequency of SR 3.6.1.1.2 should be proposed for SSES. The revised marked-up TS pages are provided for both Units 1 and 2 to replace the marked-up pages previously submitted. They are annotated to show no changes will be made on the page. Associated revised, marked-up TS Bases pages are also provided for information, to reflect that no change will be made to SR 3.6.1.1.2.

An administrative change is also necessary on the Unit 1 marked-up TS page TS/ 3.8-35. This change corrects a typographical discrepancy in the previously provided marked-up TS page for SR 3.8.6.1. The revised marked-up page is provided in Attachment 2 with the needed administrative correction. The correction of this annotation removes already deleted text, erroneously shown with a strike-out (e.g., to be deleted) annotation. The strike-out annotation had been previously incorporated from an approved amendment, and use of the image of the approved page for new marked-up annotations makes the needed correction. This does not impact the approved TS page or technically change the submittal, or existing licensing basis.

Attachment 2 to PLA-7334

Revised Marked-up TS Pages

Note: This attachment contains new marked-up TS pages to supersede the ones previously provided in letter PLA-7119, (Reference 1).

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during η Surveillance Frequency intervals, where η is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

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Marked Up Page for PLA-7334
No Changes are on this page

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.1.2	Verify that the drywell-to-suppression chamber bypass leakage is less than 0.00535 ft^2 at an initial differential pressure of ≥ 4.3 psi.	When performing 10 CFR 50 Appendix J, Type A testing, in accordance with the Primary Containment Leakage Rate Testing Program. <u>AND</u> -----------------Note---------------- Only required after two consecutive tests fail and continues until two consecutive tests pass -------------------------------------- 24 months

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No Changes are on this page

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.1.2	Verify that the drywell-to-suppression chamber bypass leakage is less than 0.00535 ft^2 at an initial differential pressure of $\geq$ 4.3 psi.	When performing 10 CFR 50 Appendix J, Type A testing, in accordance with the Primary Containment Leakage Rate Testing Program. <u>AND</u> -----------------Note---------------- Only required after two consecutive tests fail and continues until two consecutive tests pass -------------------------------------- 24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	-----------------------------------NOTE----------------------------------- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. --- Verify each battery float current is ≤ 2 amps.	~~7 days~~
SR 3.8.6.2	Verify each battery pilot cell voltage is ≥ 2.07 V.	~~31 days~~
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	~~31 days~~
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	~~31 days~~
SR 3.8.6.5	Verify each battery connected cell voltage is ≥ 2.07 V.	~~92 days~~

(continued)

In accordance with the Surveillance Frequency Control Program

Attachment 3 to PLA-7334

Revised Marked-up TS Bases Pages
(For Information)

Note: This attachment contains new marked-up TS Bases pages to supersede the ones previously provided in letter PLA-7119, (Reference 1).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 24 months is required until the situation is remedied as evidenced by passing two consecutive tests.

SR 3.6.1.1.3

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through downcomers into the suppression pool. This SR measures suppression chamber-to-drywell vacuum breaker leakage to ensure the leakage paths that would bypass the suppression pool are within allowable limits. The total allowable leakage limit is 30% of the SR 3.6.1.1.2 limit. The allowable leakage per set is 12% of the SR 3.6.1.1.2 limit.

The leakage is determined by establishing a 4.3 psi differential pressure across the drywell-to-suppression chamber vacuum breakers and verifying the leakage. ~~The leakage test is performed every 24 months. The 24 month Frequency was developed considering the surveillance must be performed during a unit outage~~. A Note is provided which allows this Surveillance not to be performed when SR 3.6.1.1.2 is performed. This is acceptable because SR 3.6.1.1.2 ensures the OPERABILITY of the pressure suppression function including the suppression chamber-to-drywell vacuum breakers. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.2.
2. FSAR, Section 15.
3. 10 CFR 50, Appendix J, Option B.
4. Nuclear Energy Institute, 94-01

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SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

The allowable limit is 10% of the acceptable SSES $A/\sqrt{k}$ design value. For SSES, the $A/\sqrt{k}$ design value is .0535 ft^2.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and determining the leakage. The leakage test is performed when the 10 CFR 50, Appendix J, Type A test is performed in accordance with the Primary Containment Leakage Rate Testing Program. This testing Frequency was developed considering this test is performed in conjunction with the Integrated Leak rate test and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 24 months is required until the situation is remedied as evidenced by passing two consecutive tests.

SR 3.6.1.1.3

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through downcomers into the suppression pool. This SR measures suppression chamber-to-drywell vacuum breaker leakage to ensure the leakage paths that would bypass the suppression pool are within allowable limits. The total allowable leakage limit is 30% of the SR 3.6.1.1.2 limit. The allowable leakage per set is 12% of the SR 3.6.1.1.2 limit.

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