PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

APR 0 9 2001

PSEG
Nuclear LLC

LR-N01-00101
LCR H00-01

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
CHANGES TO VACUUM BREAKER REQUIREMENTS
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

By letter dated October 12, 2000, PSEG Nuclear LLC submitted a request for revision to the Technical Specifications (TS) for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this letter has been sent to the State of New Jersey.

The purpose of this letter is to provide additional information and clarification to the original submittal as a result of phone conversations with Mr. R. Ennis and Mr. J. Harrison of the NRC Staff. Attachment 1 to this letter includes revisions to the October 12, 2000 letter. The changes included herein do not alter the 10CFR50.92 evaluation, with a determination of no significant hazards consideration, of the original submittal. The 10CFR50.92 evaluation is provided as Attachment 2 to this letter for the sake of completeness. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 3. These pages include a minor revision to the bases pages and one additional deletion which, although discussed in the original submittal, was inadvertently overlooked during the markup process. In addition, re-typed Technical Specification sections have been included, as Attachment 4, in order to assist with the review.

Should you have any questions regarding this request, please contact Mr. John C. Nagle at (856) 339-3171.

Sincerely,

Mark Bezilla
Vice President – Technical Support

A001

Affidavit
Attachments (4)

C Mr. H. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Ennis
Licensing Project Manager - Hope Creek
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 8B1
11555 Rockville Pike
Rockville, MD 20852

USNRC Senior Resident Inspector - HC (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P. O. Box 415
Trenton, NJ 08625

REF: LR-N01-~~000xx~~ 00101
LCR H00-01 4/10/01

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

Mark B. Bezilla, being duly sworn according to law deposes and says:

I am Vice President – Technical Support of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Hope Creek Generating Station, Unit 1, are true to the best of my knowledge, information and belief.

Mark B Bezilla

Subscribed and Sworn to before me
this April day of 9th, 2001

Kimberly J. Brown
Notary Public of New Jersey

My Commission expires on June 23, 2003

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

BACKGROUND:

The proposed changes are intended to make the Hope Creek Technical Specifications (TS) consistent with the improved standard TS (STS) contained in NUREG-1433. The submittal is based, in large degree, upon the work of the BWROG for the technical specification conversion process. These changes bring together, in one place, the requirements affecting the vacuum breakers and expand the defined actions so that potential confusion will be eliminated regarding the potential application of other appropriate containment specifications or the necessity to enter TS 3.0.3.

REQUESTED CHANGE, PURPOSE AND BACKGROUND:

For the sake of clarity change numbers from the October submittal are retained. These changes affect the proposed changes to TS 3.6.4.2. Deleted text is lined out and revisions are underlined.

1. <u>Change A10</u>: A new Action b is added to cover the condition in which two vacuum breaker assemblies have one or two valves that are inoperable for opening. <u>Unlike change L4, which discussed a single assembly, this change describes a condition with two assemblies unable to provide the pressure relief function necessary to protect the containment. This condition would appear to be undefined in the current TS; however,</u> with these conditions, primary containment integrity requirements would not be met and Hope Creek would currently default to the action of TS 3.6.1.1 that allows 1 hour for restoration. This is the same completion time as for the proposed Action b. There is therefore no change in intent and this change is considered to be administrative.

2. <u>Change A12</u>: The re-lettered Action c is modified to clarify that the action covers the condition in which one valve in each of the two vacuum breaker assemblies is not closed. <u>This revision</u> is consistent with the STS and ~~is considered to be an administrative change to provide clarification.~~ <u>describes a condition in which redundancy is lost but functionality is maintained. This change can be viewed as being less restrictive than current requirements because of the need to enter TS 3.0.3 for a condition not defined by the specifications.</u>

3. <u>Change A13</u>: The phrase “verify the other vacuum breaker assembly valve in the line to be closed within 2 hours” is deleted from the re-lettered Action c. In

accordance with TS 3.0.1, if, at any time, the other vacuum breaker assembly valve is found or known to be open, SR 4.6.4.2.a is not met and the new Action d would be entered for the upon discovery. The proposed Action d provides a more conservative action time (1 hour) than the action time in the deleted phrase (2 hours). As a result, the "verification" in the re-lettered Action c is implicitly included in the new Action d ~~and is considered to be an administrative change~~. The change is considered to be a more restrictive change because the new Action d only provides one hour to correct the condition. The deletion of the explicit requirement to check the other valve can be interpreted as being less restrictive; however, the proposed reduction in the completion time to one hour for restoring the assembly is more restrictive than the existing specification and provides for a more restrictive specification overall.

4. Change M2: A new Action d is added to cover the condition in which both valves in one or both assemblies are open. As noted above, when a vacuum breaker assembly valve is open, the current TS requires that the other assembly valve be verified closed within 2 hours. ~~Implicit in this action is the requirement to close at least one of the valves in the subject assembly within the two-hour allowance if both valves in the assembly are found open. Otherwise, the plant must be shutdown.~~ If both valves are then (within two hours) determined to be open 3.6.1.1 would apply and action would be taken to restore a valve within one hour or the plant would placed in HOT SHUTDOWN within the next 12 hours. It should be noted that both valves open may also be viewed as requiring entrance into specification 3.0.3, which has similar Actions. The action time specified in ~~The~~ new Action d decreases this time to 1 hour to be consistent with the time provided in Hope Creek TS 3.6.1.1 for primary containment integrity not maintained. ~~The reduction in the completion time is considered to be a more restrictive change~~. This is also consistent with the requirements of 3.0.3. This results in a more restrictive specification.

A comparison of Specifications 3.0.3 and 3.6.1.1 is provided for reference.

	3.0.3	3.6.1.1
Condition Not Met	T=0	T=0
Action	Action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply	restore PRIMARY CONTAINMENT INTEGRITY

Restore or Otherwise Exit LCO	T=1 hour	T=1 hour
Be in Start-Up	T=7 hours	No requirement
Be in Hot Shutdown	T=13 hours	T=13 hours
Be in Cold Shutdown	T= 37 hours	T=25 hours

The major difference in these two specifications relates to the one hour requirement. TS 3.0.3 requires that action be initiated within one hour (to shutdown the unit) whereas TS 3.6.1.1 states that the condition be corrected within one hour or the Unit be in Hot Shutdown in the subsequent 12 hours. From a practical standpoint these requirements are essentially identical. Both Actions will result in the Unit being in Hot Shutdown 13 hours from the entry into the non-conforming condition.

5. Change L8: Surveillance Requirement 4.6.4.2.b.2.b regarding visual inspection of the vacuum breaker assemblies is deleted. This paragraph was inadvertently overlooked when preparing the hand mark up and was not "lined out".

. Change to Bases: The Bases have been further modified to delete reference to the ISI program from the discussion of surveillance requirements. The current surveillance time has not changed from the existing specification and is more conservative than the ISI program allowances.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

ADMINISTRATIVE CHANGES

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

Various administrative changes are proposed for TS 3.6.4.1, 3.6.4.2, and 3.6.2.1. These administrative changes are addressed by the evaluation provided below.

BASIS

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

The proposed changes are administrative in nature and do not involve any technical changes. These changes to the Hope Creek TS are being made to in order to provide consistency between the Hope Creek TS and the improved standard TS (NUREG-1433). Being administrative in nature, these changes do not impact accident initiators, analyzed events, or the mitigation of accidents or transients. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

MORE RESTRICTIVE CHANGES

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

Various more restrictive changes are proposed for TS 3.6.4.1 and 3.6.4.2. These more restrictive changes are addressed by the evaluation provided below.

BASIS

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

The proposed changes provide more restrictive requirements than previously existed in the TS. The more restrictive requirements will not result in operation that will increase the probability of initiating an analyzed event. The new requirements either do not change or, in some instances, may decrease the probability or consequences of an analyzed event. These changes will not invalidate assumptions relative to mitigation of an accident or transient event. These changes have been reviewed to ensure that no previous accident evaluations have been adversely impacted. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The more restrictive requirements imposed by the proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed). Any resulting changes in the methods governing plant operation will be consistent with assumptions made in the safety analyses. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The more restrictive requirements imposed by the proposed changes either increase or do not affect the margin of safety. In addition, the proposed changes do not impact any safety analysis assumptions. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

CHANGES INVOLVING RELOCATION OF REQUIREMENTS

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

The proposed changes relocate requirements from TS 3.6.4.1 and 3.6.4.2 to licensee-controlled documents. These changes that involve relocation of requirements are addressed by the evaluation provided below.

BASIS

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

The proposed changes relocate requirements from the TS to licensee-controlled documents. Any future changes to the licensee-controlled documents containing relocated requirements will be evaluated in accordance with the PSEG Nuclear 10CFR50.59 program. Since any changes to licensee-controlled documents will be evaluated in accordance with the PSEG Nuclear 10CFR50.59 program, no increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC approval. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed changes relocate requirements from the TS to licensee-controlled documents. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Adequate control of these requirements will be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes relocate requirements from the TS to licensee-controlled documents. The proposed changes will not reduce a margin of safety since the changes have no impact on any safety analysis assumptions. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Since any changes to licensee-controlled documents will be evaluated in accordance with the PSEG Nuclear 10CFR50.59 program, no reduction in a margin of safety will be allowed without prior NRC approval. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

CHANGES INVOLVING DELETION OF POSITION INDICATION AND ACTUATION SYSTEM REQUIREMENTS

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

Deletion of Hope Creek TS 3.6.4.1 Action c and TS 3.6.4.2 Action c regarding the vacuum breaker position indicators, and the associated surveillance requirements (SR 4.6.4.1.b.2, 4.6.4.1.b.3.b, 4.6.4.2.b.1.b, and 4.6.4.2.b.2.c) is proposed. Deletion of SR 4.6.4.2.b.2.d regarding verification of the instrument actuation system for the inboard isolation valve auto open control system operability by channel calibration is also proposed. These changes are addressed by the evaluation provided below.

BASIS

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

The proposed changes do not result in any hardware or operating procedure changes. The vacuum breaker position indication or actuation system instrumentation are not assumed in the initiation of any analyzed event. The requirements for the vacuum breaker position indication or actuation system instrumentation do not need to be explicitly stated in the TS. The capability to determine vacuum breaker position and the vacuum breaker actuation instrumentation must be available to perform the verifications and tests required for the surveillance requirements of the TS. If the capability to determine vacuum breaker position is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the actions of the TS. As a result, accident consequences are unaffected by the proposed change. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed changes do not introduce any new modes of plant operation or involve a physical alteration of the plant. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed deletion of the vacuum breaker position indication and actuation instrumentation requirements from the TS does not impact a margin of safety. The requirements for the vacuum breaker position indication and actuation instrumentation do not need to be explicitly stated in the TS. The capability to determine vacuum breaker position and the vacuum breaker actuation instrumentation must be available to perform the verifications and tests required for the surveillance requirements of the TS. If the capability to determine vacuum breaker position and the vacuum breaker actuation instrumentation is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the actions of the TS. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

CHANGES INVOLVING REDUCTION IN FREQUENCY OF CLOSED POSITION VERIFICATION

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

The frequency of verifying that each vacuum breaker is closed is changed from once per 7 days to once per 14 days in Surveillance Requirements 4.6.4.1.a and 4.6.4.2.a. These changes are addressed by the evaluation provided below.

BASIS

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

The proposed changes would decrease the surveillance frequency of the vacuum breaker position verification from once per 7 days to once per 14 days. The proposed change does not affect the vacuum breaker valve design or function. A failure of a vacuum breaker valve is not identified as an initiator of any event. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated. Since the change impacts only the frequency of verification and does not result in any change in the response of the equipment to an accident, the change does not increase the consequences of any previously analyzed accident. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed changes do not result in any changes to the equipment design or capabilities or to the operation of the plant. The proposed changes impact only the frequency of verification of vacuum breaker position and do not result in any change in the response of equipment to an accident. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes impact only the frequency of verification of the vacuum breaker position. Hope Creek experience has shown that a change to 14 days to verify that a vacuum breaker is closed is not a significant change in operating practice and that the proposed frequency is acceptable. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

CHANGES INVOLVING EXTENSION OF TIME FOR PERFORMING FUNCTIONAL TESTING FOLLOWING STEAM DISCHARGE

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

The time requirement to perform functional testing after any discharge of steam to the suppression chamber from the safety relief valves (SRVs) is changed from 2 hours to 12 hours. This change is addressed by the evaluation provided below.

BASIS

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

The proposed change extends the requirement to cycle the vacuum breakers after an SRV lift from 2 hours to 12 hours. Since the vacuum breakers are not assumed to be an initiator of any previously analyzed accident, the change will not significantly increase the probability of a previously analyzed accident. Since sufficient vacuum breakers will remain operable to mitigate the assumed accidents, the change will not increase the consequences of a previously analyzed accident. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The operability of the vacuum breakers is not adversely affected by an SRV lift. Since the vacuum breakers are designed to operate and assumed to function after a LOCA blowdown, operation of the vacuum breakers following a minor steam release from the SRVs should not raise any questions regarding immediate operability. Steam discharged to the suppression chamber, resulting in increased pressure and vacuum breaker opening, could pose a long-term equipment degradation issue, but not an immediate operability concern, therefore the potential impact from the proposed ten hour increase is minimal. In addition, the basis for this extension is also supported by NRC Generic Letter 93-05, Item 8.4. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

INCLUSION OF CONDITION GOVERNING BOTH VALVES IN A REACTOR BUILDING –SUPPRESSION CHAMBER VACUUM BREAKER ASSEMBLY BEING INOPERABLE FOR OPENING

<u>10CFR50.92 EVALUATION</u>

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

Action statement 3.6.4.2.a is modified to include the condition in which both valves in one vacuum breaker assembly are inoperable for opening. These changes are addressed by the evaluation provided below.

BASIS

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

The existing action includes only the condition in which one valve in an assembly is inoperable for opening, while the proposed change recognizes that there are two valves in series in each of two vacuum breaker assemblies between the reactor building and suppression chamber. The proposed change will make a distinction between loss of function (containment integrity and venting capability) that requires initiating action within one hour (both valves inoperable for opening) and loss of redundancy for a function that must be recovered within 72 hours (one valve in the assembly is inoperable for opening). The existing TS fails to make this distinction between loss of function and loss of redundancy. The probability of an accident is not increased because the vacuum breakers are not considered to be the initiators of any accidents previously evaluated. The consequences of an accident will not be increased because the proposed change will provide assurance that both the containment integrity and venting capability functions are available or restored within one hour. The proposed change could allow continued operation for up to 72 hours without redundant capability for these functions; however, the 72 hour completion time accounts for the redundant capability provided by the remaining vacuum breaker assembly, the fact that the operable assembly is closed, and the low probability of an event that would require the vacuum breaker valves to be operable during this period (consistent with the existing action statement). Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed change does not result in any physical changes to plant systems, structures, or components (SSCs) or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will make a distinction between loss of function (containment integrity and venting capability) that requires initiating action within one hour and loss of redundancy for a function that must be recovered within 72 hours. The existing TS fails to make this distinction between loss of function and loss of redundancy. The proposed change will provide assurance that both the containment integrity and venting capability functions are available or restored within one hour. The change does not affect the current analysis assumptions. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

DELETION OF SURVEILLANCE REGARDING VISUSAL INSPECTION OF VACUUM BREAKERS

10CFR50.92 EVALUATION

PSEG Nuclear LLC has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGES

Surveillance Requirement 4.6.4.2.b.2.b regarding visual inspection of the vacuum breaker assemblies is deleted. This change is addressed by the evaluation provided below.

BASIS

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

The proposed change does not result in any hardware or operating procedure changes. The vacuum breakers are not assumed in the initiation of any analyzed event. The requirements for the vacuum breaker visual inspections do not need to be explicitly stated in the TS. The performance of the verifications and tests required for the Surveillance Requirements of this TS and the proposed SR 4.6.2.1.f ensures the operability of the vacuum breakers. As a result, accident consequences are unaffected by this change. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. ***The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.***

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The requirements for the vacuum breaker visual inspections do not need to be explicitly stated in the TS. The performance of the verifications and tests required for the Surveillance Requirements of this TS and the proposed SR 4.6.2.1.f ensures the operability of the vacuum breakers. As a result, operability of the vacuum breakers will be maintained without the need for explicit visual inspection requirements. Therefore, the changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
4.6.2.1	3/4 6-13 and 3/4 6-14
3/4.6.4.1	3/4 6-43 and 3/4 6-44
3/4.6.4.2	3/4 6-45 and 3/4 6-46
Bases 3/4.6.4	B 3/4 6-5

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER.

c. ~~c)~~ At least once per 30 minutes in OPERATIONAL CONDITION 3 following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

d. ~~c.~~ By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 177°F and reactor coolant system pressure greater than 100 psig.

e. ~~d.~~ At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

f. ~~e.~~ At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 0.80 psi and verifying that the differential pressure does not decrease by more than 0.24 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be OPERABLE ~~and closed~~. e

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above vacuum breakers inoperable for opening ~~but known to be closed~~, restore the ~~inoperable~~ vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. e

b. With one ~~or more~~ e suppression chamber - drywell vacuum breaker~~(s)~~ e ~~open~~, not closed close the open vacuum breaker~~(s)~~ within 2 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. ~~With one of the position indicators of any suppression chamber - drywell vacuum breaker inoperable:~~ e

1. ~~Verify the other position indicator in the pair to be OPERABLE within 2 hours and at least once per 14 days thereafter, or~~

2. ~~Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 14 days thereafter.~~

~~Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per ~~7~~ 14 days.*

b. Demonstrated OPERABLE:

1. At least once per 31 days and within ~~2~~ 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by ~~cycling each vacuum breaker through at least one complete cycle of full travel.~~ performing a functional test of each vacuum breaker.

~~2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.~~

2 ~~3.~~ At least once per 18 months by ~~a)~~ verifying the opening setpoint of each vacuum breaker ~~from the closed position,~~ to be less than or equal to 0.20 psid. ~~and~~

~~b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.~~

* Not required to be met for vacuum breaker assembly valves that are open during surveillances or that are open when performing their intended function

CONTAINMENT SYSTEMS

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 EACH ~~Both~~ reactor building - suppression chamber vacuum breaker assemb~~lies~~ y ~~consisting of a vacuum breaker valve and a butterfly isolation valve~~ shall be OPERABLE ~~and closed.*~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one ~~valve of a~~ reactor building - suppression chamber vacuum breaker assembly with one or two valves inoperable for opening ~~but known to be closed~~, restore the ~~inoperable~~ vacuum breaker assembly ~~valve~~ to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Insert A

~~b.~~ C. With one or two ~~valve of a~~ reactor building - suppression chamber vacuum breaker assembl~~y open~~ ies with one valve not closed, ~~verify the other vacuum breaker assembly valve in the line to be closed within 2 hours; restore~~ close the open vacuum breaker assembly valve(s) ~~to the closed position~~ within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~c. With the position indicator of any reactor building - suppression chamber vacuum breaker assembly valve inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the affected vacuum breaker assembly valve to be closed at least once per 24 hours by a visual inspection. Otherwise, declare the vacuum breaker assembly valve inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

Insert B

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each ~~Both~~ reactor building - suppression chamber vacuum breaker assembl~~ies~~ y shall be:

a. Verified closed at least once per ~~7~~ 14 days.*

b. Demonstrated OPERABLE:

1. At least once per 31 days by: Performing a functional test of each vacuum breaker assembly valve.

   a) ~~Cycling each vacuum breaker assembly valve through at least one complete cycle of full travel.~~

   ~~b) Verifying the position indicators on each assembly valve OPERABLE by observing expected valve movement during the cycling test.~~

2. At least once per 18 months by: Verifying the opening setpoint of

   a) ~~Demonstrating that the force required to open~~ each vacuum breaker assembly valve ~~does not exceed the equivalent of~~ to be less than or equal to 0.25 psid.

   ~~b) Visual inspection.~~

* Not required to be met for vacuum breaker assembly valves that are open during surveillances or that open when performing their intended function.

Insert A

b. With two reactor building - suppression chamber vacuum breaker assemblies with one or two valves inoperable for opening, restore both valves in one vacuum breaker assembly to OPERABLE status within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Insert B

d. With two valves in one or two reactor building – suppression chamber vacuum breaker assemblies not closed, close one open vacuum breaker assembly valve in each affected assembly within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Insert A

b. With two reactor building - suppression chamber vacuum breaker assemblies with one or two valves inoperable for opening, restore both valves in one vacuum breaker assembly to OPERABLE status within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Insert B

d. With two valves in one or two reactor building – suppression chamber vacuum breaker assemblies not closed, close one open vacuum breaker assembly valve in each affected assembly within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

~~c) Verifying the position indicators on each assembly valve OPERABLE by performance of a CHANNEL CALIBRATION.~~

~~d) Verifying the instrument actuation system for the inboard isolation valve auto open control system OPERABLE by performance of a CHANNEL CALIBRATION.~~

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 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 C

~~Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. 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.~~

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a 0.25 inch water gage vacuum in the reactor building with the filtration recirculation and ventilation system (FRVS) once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the FRVS ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses and with the drawdown analysis. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Insert C (Replaces Insert C from Oct submittal)

Suppression Chamber-to-Drywell Vacuum Breakers

BACKGROUND: The function of the suppression-chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are eight internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber that allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the wetwell-drywell boundary. Each vacuum breaker is a self-actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, inadvertent drywell spray actuation, and steam condensation from sprays or subcooled water reflood of a break in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or spill of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, air in the drywell is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused by Emergency Core Cooling Systems flow from a recirculation line or main steam line break, or drywell spray actuation following a loss of coolant accident (LOCA).

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is less than the suppression chamber pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.

APPLICABLE SAFETY ANALYSES: Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Section 6.2 and Appendix 6A of the Hope Creek UFSAR as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building- to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary.

The safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.20 psid. Additionally, one of the eight internal vacuum breakers is assumed to fail in a closed position. The results of the analyses show that the design pressure limits are not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that all eight vacuum breakers be OPERABLE are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The vacuum relief capacity between the drywell and suppression chamber should be 1/16 of the total main vent cross sectional area, with the valves set to operate at 0.20 psid differential pressure. Design Basis Accident (DBA) analyses require the vacuum breakers to be closed initially and to remain closed and leak tight.

The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of the NRC Policy Statement.

LCO: All eight vacuum breakers must be OPERABLE for opening and closed (except during testing or when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY: In OPERATIONAL CONDITIONS 1, 2, and 3, the Suppression Pool Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the drywell could occur due to inadvertent actuation of this system. The vacuum breakers, therefore, are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System.

Also, in OPERATIONAL CONDITIONS 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in OPERATIONAL CONDITIONS 1, 2, and 3.

In OPERATIONAL CONDITIONS 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these OPERATIONAL CONDITIONS; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in OPERATIONAL CONDITION 4 or 5.

ACTIONS: With one of the required vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining seven OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one of the eight required vacuum breakers inoperable, 72 hours is allowed to restore at least one of the inoperable vacuum breakers to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event and the adequacy of the remaining vacuum breaker capability.

An open vacuum breaker allows communication between the drywell and suppression chamber airspace, and, as a result, there is the potential for suppression chamber overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is to verify that a differential pressure of 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The required 2 hour Completion Time is considered adequate to perform this test.

If the inoperable suppression chamber-to-drywell vacuum breaker cannot be closed or restored to OPERABLE status within the required Completion Time, the plant must be brought to an OPERATIONAL CONDITION in which the LCO does not apply. To achieve this status, the plant must be brought to at least OPERATIONAL CONDITION 3 within 12 hours and to OPERATIONAL CONDITION 4 within the following 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS: Each vacuum breaker is verified closed to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that a differential pressure of 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

A Note is added to this SR that allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The 31-day Frequency of this SR was chosen to provide additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after a discharge of steam to the suppression chamber from the safety/relief valves.

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.20 psid is valid. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this facility, the 18-month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

Reactor Building-to-Suppression Chamber Vacuum Breakers

BACKGROUND: The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression-chamber-to-drywell vacuum breakers. The design of the external (reactor building-to-suppression chamber) vacuum relief provisions consists of two vacuum breakers (a check type vacuum relief valve and an air operated butterfly valve located in series) in each of two lines from the reactor building to the suppression chamber airspace. The butterfly valve is actuated by differential pressure. The vacuum breaker is self-actuating and can be remotely operated for testing purposes. The two vacuum breakers in series must be closed to maintain a leak tight primary containment boundary.

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, inadvertent primary containment spray actuation, and steam condensation in the event of a primary system rupture. Reactor building-to-suppression chamber vacuum breakers prevent an excessive negative differential pressure across the primary containment boundary. Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by heating and ventilation equipment. Inadvertent spray actuation results in a more significant pressure transient and becomes important in sizing the external (reactor building-to-suppression chamber) vacuum breakers.

The external vacuum breakers are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere. Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

<u>APPLICABLE SAFETY ANALYSES</u>: Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Section 6.2 and Appendix 6A of the Hope Creek UFSAR as part of the accident response of the containment systems. Internal (suppression-chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 0.25 psid. Additionally, of the two reactor building-to-suppression chamber vacuum breakers, one is assumed to fail in a closed position to satisfy the single active failure criterion. Design Basis Accident (DBA) analyses require the vacuum breakers to be closed initially and to remain closed and leak tight with positive primary containment pressure.

The reactor building-to-suppression chamber vacuum breakers satisfy Criterion 3 of the NRC Policy Statement.

LCO: All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two vacuum breakers (vacuum breaker and air operated butterfly valve) in each of the two lines from the reactor building to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber.

APPLICABILITY: In OPERATIONAL CONDITIONS 1, 2, and 3, a DBA could cause pressurization of primary containment. In OPERATIONAL CONDITIONS 1, 2, and 3, the Suppression Pool Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside primary containment could occur due to inadvertent initiation of this system. Therefore, the vacuum breakers are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System.

Also, in OPERATIONAL CONDITIONS 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in OPERATIONAL CONDITIONS 1, 2, and 3.

In OPERATIONAL CONDITIONS 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these OPERATIONAL CONDITIONS. Therefore, maintaining reactor building-to-suppression chamber vacuum breakers OPERABLE is not required in OPERATIONAL CONDITION 4 or 5.

ACTIONS: Action a: With one vacuum breaker assembly with one or two valves inoperable for opening, the leak tight primary containment boundary is intact. The ability to mitigate an event that causes a containment depressurization is threatened, however, if both vacuum breakers in at least one vacuum breaker assembly are not OPERABLE. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status within 72 hours. This is consistent with the Completion Time for Action c and the fact that the leak tight primary containment boundary is being maintained.

Action b: With two vacuum breaker assemblies with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the function of the vacuum breakers is lost. Therefore, both valves in one assembly must be restored to OPERABLE status within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

Action c: With one or more vacuum breaker assemblies with one valve not closed, the leak tight primary containment boundary may be threatened. Therefore, the inoperable valves must be restored to OPERABLE status or the open vacuum breaker assembly valve closed within 72 hours. The 72 hour Completion Time is consistent with requirements for inoperable suppression-chamber-to-drywell vacuum breakers in LCO 3.6.4.1, "Suppression-Chamber-to-Drywell Vacuum Breakers." The 72 hour Completion Time takes into account the redundant capability afforded by the remaining valves, the fact that an OPERABLE valve in each of the assemblies is closed, and the low probability of an event occurring that would require the valves to be OPERABLE during this period.

Action d: With one or more vacuum breaker assemblies with two valves not closed, primary containment integrity is not maintained. Therefore, one open valve in each affected assembly must be closed within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

If all the valves in a vacuum breaker assembly cannot be closed or restored to OPERABLE status within the required Completion Time, the plant must be brought to an OPERATIONAL CONDITION in which the LCO does not apply. To achieve this status, the plant must be brought to at least OPERATIONAL CONDITION 3 within 12 hours and to OPERATIONAL CONDITION 4 within the following 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

<u>SURVEILLANCE REQUIREMENTS</u>: Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

A Note is added to this SR. The first part of the Note allows reactor-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second part of the Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR is more conservative than the Inservice Testing Program requirements.

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.25 psid is valid. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this unit, the 18 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

The revised Technical Specification Sections are retyped below to assist in the review. Camera ready pages will be provided at the request of NRC.

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above vacuum breakers inoperable for opening, restore the vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker not closed, close the open vacuum breaker within 2 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 14 days*.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by performing a functional test of each vacuum breaker.

2. At least once per 18 months by verifying the opening setpoint, for each vacuum breaker to be less than or equal to 0.20 psid.

*Not required to be met for vacuum breaker assembly valves that are open during surveillances or that are open when performing their intended functions.

3.6.4.2 Each reactor building - suppression chamber vacuum breaker assembly shall be OPERABLE.

a. With one reactor building - suppression chamber vacuum breaker assembly, with one or two valves inoperable for opening, restore the vacuum breaker assembly to OPERABLE status within 72 hours or be in at least HOT

SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two reactor building - suppression chamber vacuum breaker assemblies with one or two valves inoperable for opening, restore both valves in one vacuum breaker assembly to OPERABLE status within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one or two reactor building - suppression chamber vacuum breaker assemblies, with one valve not closed, close the open vacuum breaker assembly valve(s) within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two valves in one or two reactor building – suppression chamber vacuum breaker assemblies not closed, close one open vacuum breaker assembly valve in each affected assembly within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6.4.2 Each reactor building - suppression chamber vacuum breaker assembly shall be:

a. Verified closed at least once per 14* days.

b. Demonstrated OPERABLE:

1. At least once per 31 days by:

a) Performing a functional test of each vacuum breaker assembly valve.

2. At least once per 18 months by:

a) Verifying the opening setpoint of each vacuum breaker assembly valve to be less than or equal to 0.25 psid.

*Not required to be met for vacuum breaker assembly valves that are open during surveillances or that are open when performing their intended functions.