Mr. Oliver D. Kingsle Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

July 8, 1997

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 - REVISION TO TECHNICAL SPECIFICATION BASES (TAC NOS. M97911, M97912, M97913, M98695, AND M98696) (TS 388 AND TS 389)

Dear Mr. Kingsley:

By letters dated February 5 and April 24, 1997, the Tennessee Valley Authority provided changes to the Technical Specification (TS) Bases for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The February 5, 1997 Bases revisions describe an alternate method for drywell to suppression chamber leak testing for BFN Units 1, 2, and 3. The April 24, 1997 Bases revisions add references for updated loss of coolant accident analyses for BFN Units 2 and 3.

The enclosures list the appropriate BFN Units 1, 2, and 3 TS Bases modified pages.

Please call me at (301)415-1470 if you have any questions regarding this topic.

Sincerely,

Original signed by

Joseph F. Williams, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosures: TS Bases Pages

cc w/Enclosures: See next page

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Mr. Oliver D. Kingsley, Jr.
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

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Chairman
Limestone County Commission
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Athens, AL 35611

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, AL 36130-1701

REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 1

The revised page contains marginal lines indicating the areas of change.
*Overleaf page is provided to maintain document completeness.

Remove	Insert
3.7/4.7-29	3.7/4.7-29
3.7/4.7-30	3.7/4.7-30*

confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully OPERABLE. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure unless all vacuum breakers can be demonstrated to be fully closed by alternate means such as monitoring the decay rate of drywell to suppression chamber differential pressure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered OPERABLE the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten OPERABLE to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 2

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Remove	Insert
3.5/4.5-32	3.5/4.5-32
3.5/4.5-33	3.5/4.5-33
3.5/4.5-34	3.5/4.5-34
3.5/4.5-35	3.5/4.5-35*
3.7/4.7-29	3.7/4.7-29
3.7/4.7-30	3.7/4.7-30*

3.5 BASES (Cont'd)

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. <u>Minimum Critical Power Ratio (MCPR)</u>

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. <u>APRM Setpoints</u>

Operation is constrained to the LHGR limit of Specification 3.5.J. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A six-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. <u>Core Thermal-Hydraulic Stability</u>

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Bases change 7/8/97

3.5 BASES (Cont'd)

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.

4. GE Document NEDC-32484P, Rev. 1, S. K. Rhow and C. T. Young, "Browns Ferry Nuclear Plant Units 1, 2, and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," February 1996.

5. GE Document GE-NE-B13-01755-2, Rev. 1, S. K. Rhow and T. H. Chuang, "Relaxation of Emergency Core Cooling System Parameters for Browns Ferry Nuclear Plant Units 1, 2, and 3 (Perform Program Phase I)," February 1996.

Bases change 7/8/97

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

Basis change 7/8/97

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confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

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Bases Change 7/8/97

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REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 3

The revised pages contain marginal lines indicating the areas of change.
*Overleaf pages are provided to maintain document completeness.

Remove	Insert
3.5/4.5-35	3.5/4.5-35
3.5/4.5-36	3.5/4.5-36
3.5/4.5-37	3.5/4.5-37
3.5/4.5-38	3.5/4.5-38*
3.7/4.7-28	3.7/4.7-28
3.7/4.7-29	3.7/4.7-29*

3.5 BASES (Cont'd)

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. APRM Setpoints

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Bases change 7/8/97

3.5 BASES (Cont'd)

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Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

4. GE Document NEDC-32484P, Rev. 1, S. K. Rhow and C. T. Young, "Browns Ferry Nuclear Plant Units 1, 2, and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," February 1996.

5. GE Document GE-NE-B13-01755-2, Rev. 2, S. K. Rhow and T. H. Chuang, "Relaxation of Emergency Core Cooling System Parameters for Browns Ferry Nuclear Plant Units 1, 2, and 3 (Perform Program Phase I)," December 1996.

Basis change 7/8/97

4.5 <u>Core and Containment Cooling Systems Surveillance Frequencies</u>

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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<u>Average Planar LHGR, LHGR, and MCPR</u>

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

Basis change 7/8/97

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The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

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Basis change 7/8/97

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this Bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully OPERABLE. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure unless all vacuum breakers can be demonstrated to be fully closed by alternate means such as monitoring the decay rate of drywell to suppression chamber differential pressure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered OPERABLE the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten OPERABLE to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

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With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

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