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PSEG
Nuclear LLC

APR 1 1 2001

LRN-01-00102
LCR H00-06
RR V-005

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

Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS AND RELIEF REQUEST
EXCESS FLOW CHECK VALVE TESTING REQUIREMENTS
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

In accordance with 10CFR50.90 and 10CFR50.55a, PSEG Nuclear LLC hereby requests a revision to the Technical Specifications (TS) and requests relief from the requirements of the Inservice Testing Program for the Hope Creek Generating Station (HC). In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed change relaxes the TS requirements for testing of excess flow check valves for Hope Creek. Similar changes were approved by the NRC for the following Units with Safety Evaluation Reports dated as indicated: Duane Arnold Energy Center, Unit 1, December 29, 1999, Fermi Unit 2, March 14, 2000, Browns Ferry Units 2 and 3, January 29, 2001, Limerick Generating Station Units 1 and 2, February 23, 2001 and Columbia Generating Station, February 20, 2001.

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and a determination has been made that this request involves no significant hazards considerations. The basis for the requested change is provided in Attachment 1 to this letter. A 10CFR50.92 evaluation, with a determination of no significant hazards consideration, is provided in Attachment 2. Relief request V-005 associated with these changes is provided in Attachment 3. The marked-up Technical Specification pages affected by the proposed changes are provided in Attachment 4. The proposed Bases changes are provided for information only in Attachment 5

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PSEG currently plans to implement the proposed changes in the upcoming outage scheduled for October 2001. Therefore PSEG requests that the NRC approve of this proposed change by September 1, 2001 in order to support our scheduled implementation date.

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

Sincerely,

D. Garchow
Vice President - Operations

Affidavit
Attachments (5)

C Mr. H. Miller, Administrator - Region I
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USNRC Senior Resident Inspector - HC (X24)

Mr. K. Tosch, Manager IV
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REF: LRN-01-00102
LCR H00-06
RR V-005

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

D. Garchow, being duly sworn according to law deposes and says:

I am Vice President – Operations 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.

____

Subscribed and Sworn to before me
this 17th day of April, 2001

____
Notary Public of New Jersey

My Commission expires on 6/23/2003

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
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REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)
REVISIONS TO INSERVICE TEST PROGRAM

DESCRIPTION OF THE REQUESTED CHANGE:

PSEG Nuclear is requesting a change to the Hope Creek Technical Specifications (TS) that will relax the testing frequency of the excess flow check valves (EFCV) by allowing a representative sample (approximately 15%) to be tested every 18 months such that each EFCV is tested at least once every 120 months. The current TS Surveillance Requirement (4.6.3.4) states that each reactor instrumentation line EFCV be actuated to the isolation position on a simulated instrument line break every 18 months.

The proposed change to the TS is described below. Attachment 4 contains copies of the appropriate marked-up TS pages for Hope Creek. In addition, Attachment 5 provides the associated marked-up TS Bases pages. These are provided to the NRC for information only.

Hope Creek TS, page 3/4 6-18, Primary Containment Isolation Valves, Surveillance Requirement 4.6.3.4 will be revised by adding requirements that will allow testing of a representative sample of the EFCV's on an 18 month interval. The revised surveillance requirement will read as follows:

4.6.3.4 At least once per 18 months, verify that a representative sample of reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 actuates to the isolation position on a simulated instrument line break signal.

PURPOSE OF THE REQUESTED CHANGES:

PSEG Nuclear is requesting a change to the Hope Creek TS that will reduce the number of EFCV's tested as part of refueling outage activities and online operations. Instrument line EFCV's which connect to the reactor coolant pressure boundary are normally tested during the reactor pressure vessel system leakage test, This testing is performed as part of refueling outage activities, or online with the reactor at normal operating pressure just prior to the outage (as described in the Safety Evaluation, dated December 21, 1999, for Hope Creek In-service Testing Program Relief Request No. V04).

The EFCV's are designed to close upon failure of an instrument line downstream of the valve. Testing of the EFCV's typically requires that the reactor be pressurized to normal operating pressure, opening an instrument drain valve and observing valve closure. Testing has historically required approximately 450 man-hours to complete and can be a critical path item during refueling outages. This places an undue burden on the PSEG Nuclear staff without a commensurate increase in plant safety.

Reducing the number of valves tested, as with any reduction in maintenance, inherently reduces the risk of industrial and occupational hazards, including inadvertent exposure to radioactive fluids. In addition, the availability of the associated instruments will be increased. Furthermore, there is a consequential reduction in radioactive waste generated during testing activities.

JUSTIFICATION OF REQUESTED CHANGES:

The safety objective of the Primary Containment Isolation System is to provide the capability, in the event of the postulated loss-of-coolant accident, to limit the release of fission products to the plant environs so that offsite and control room doses would be within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. Isolation of all pipes or ducts that penetrate the primary containment is required to maintain leakage within permissible limits.

The proposed change would allow a representative sample of EFCV's to be tested each 18 months. In that way, each EFCV would be tested at least once every 120 months (nominal). This test program is consistent with that described in General Electric NEDO-32977-A (Boiling Water Reactor Owners Group (BWROG) Topical Report B21-00658-01), Excess Flow Check Valve Testing Relaxation, dated November 1998, (revised through June 2000). The NRC staff approved this report on March 14, 2000.

Periodic testing of a representative sample of EFCV's, selected on a performance basis, will continue to ensure the reliability of these valves. This, along with the plant design, assures that the assumed release rate in the plant's safety analysis remains conservative.

BWROG Topical Report B21-00658-0I (subsequently re-designated as NEDO 32977-A) was reviewed, along with the licensing requirements, operational experience, and consequences associated with the testing requirements for EFCV's in instrument lines connecting to the reactor coolant pressure boundary. The report concluded that the change in the test frequency had insignificant impact on valve reliability. The BWROG report also concluded that the demonstrated reliability of EFCV's, coupled with low consequences of EFCV failure, provided adequate justification for extending the test interval up to once every 120 months.

Failure of an EFCV to close does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrumentation piping connected to the reactor primary system, which leaves the primary containment, is dead ended at instrument racks located in the Reactor Building. These instrument lines are provided with a manual block valve and an EFCV both of which are located outside primary containment. Except for the jet pump sensing lines, a one fourth (1/4)-inch orifice has been installed in instrument lines that penetrate the primary containment boundary into the secondary containment (Reactor Building area). Sensing lines for jet pump flow within the reactor vessel, including the reactor vessel penetration, are constructed of one fourth (1/4)-inch sensing pipe. The one fourth (1/4)-inch sense line, effectively, provides the same flow area as the one fourth (1/4)-inch orifice in the other instrument lines. This design limits the release of reactor coolant in the event of an instrument line break outside primary containment.

The leakage from a postulated broken instrument line outside containment is reduced by design to the maximum extent practical, consistent with instrument response requirements. The rate and extent of coolant loss is well within the capability of reactor coolant make-up systems. The integrity and functional performance of secondary containment and the Filtration, Recirculation and Ventilation System (FRVS) will be maintained. The potential exposure will be substantially below the limits of 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19. A break in the portion of an instrument line between the containment and EFCV located outside primary containment and the direct blow down to the reactor building was considered. It was concluded in the October 1984, Safety Evaluation for Hope Creek (NUREG-1048), that the isolation provisions for instrument lines penetrating primary containment were adequately designed and met the intent of NRC Regulatory Guide 1.11.

The effect of extending the EFCV test interval is a corresponding increase in the potential for a release. However, even with a 120 month test interval, the release frequency from an individual line continues to remain very low. Also, since the EFCV's are located in secondary containment (Reactor Building), any release from a failed EFCV would be treated by the Filtration Recirculation Ventilation System (FRVS) providing additional mitigation of any postulated offsite release from a broken instrument line.

The reliability of EFCV's were evaluated in the BWROG report (NEDO 32977-A) that forms the basis for this request. The composite data based on testing experience provided by 12 different BWR plants indicated that EFCV's are very reliable. A review of the maintenance history of the excess flow check valves at Hope Creek supports this conclusion. Functional testing of valves to verify closure can be accomplished by the process of venting the instrument side of the valve while the process side is under pressure. Such testing is required by Technical Specification 4.6.3.4 at least once per 18 months. Hope Creek system design does not include test taps upstream of the EFCV's. For this reason, the EFCV's cannot be isolated and tested using a pressure source other than reactor pressure.

A review of the maintenance history for EFCV's has shown that the valves have been extremely reliable over the life of the plant. Examples of causes for test failures included alarm problems, indication (limit switch adjustments), and bent instrument tubing. There has been only one failure of a valve to check flow since the initial start up of the plant and this failure resulted in the replacement of only one valve. This review of the surveillance test history shows no evidence of time based failure mechanisms or common mode failures associated with the EFCV's during Hope Creek's reactor critical operations in excess of 106,000 hours. These results are consistent with industry experience.

Any future EFCV failure would be evaluated per PSEG Corrective Action Program. Additionally, as part of implementation of this TS amendment, the 10 CFR 50.65 Maintenance Rule Program, will be revised to include a specific EFCV performance acceptance criteria.

Release Frequency

The calculations contained in the BWROG report utilize the results of surveillance testing at 12 BWR plants. These results represent a total of 12,424.5 valve operating years with a plant average of 1035 valve years per plant. There were 11 reported EFCV failures during this period, resulting in a composite failure rate of 1.01E-7/hr. At Hope Creek there was one EFCV failure in over 15 years of testing experience for 106 valves (1590 valve operating years), resulting in a failure rate of 7.2E-8/hr. The Hope Creek data is consistent both in service time sampled, and reliability, with the results listed in the BWROG report. Therefore, we have concluded that the report bounds the reliability of Hope Creek's EFCVs.

Conclusions

Implementation of this change represents an insignificant increase in release frequency, especially since any postulated coolant leakage is within the capability of the reactor coolant makeup systems, and the consequences of such an accident are not expected to lead to a core damage event.

The radiological consequences from an instrument line break were found to be a very small portion of the 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19 limits. In the unlikely event that an instrument line breaks and the EFCV fails to close, core damage would not be expected to occur and the doses would be less than regulatory limits.

Therefore, it has been concluded, considering the low consequences of a release, the extension of the test interval does not significantly affect the risk to the public associated with the failure of an instrument line and the failure of an EFCV to perform its intended function.

References:

General Electric NEDO-32977-A (Boiling Water Reactor Owners Group (BWROG) Topical Report B21-00658-01), Excess Flow Check Valve Testing Relaxation, dated November 1998, (revised through June 2000)
Technical Specification Task Force (TSTF) Item Number-334 Rev 2, "Relaxed Frequency for Excess Flow Check Valve Testing" approved by the NRC September 18, 2000
NRC SER for B21-00658-01 March 14, 2000 forwarded by letter Stuart Richards, NRC to G Warren BWROG.

ENVIRONMENTAL IMPACT:

The proposed TS changes were reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in the individual or cumulative occupational radiation exposures. Based on the foregoing, PSE&G concludes that the proposed TS changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

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10CFR50.92 EVALUATION

PSEG Nuclear has concluded that the proposed changes to the Hope Creek Generating Station (HC) Technical Specifications 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

This proposed change to Hope Creek Technical Specification 4.6.3.4 revises required testing of excess flow check valves (EFCV's) from once per 18 months for all valves to a test of a representative sample each 18 months such that all valves are tested once in ten years.

The Boiling Water Reactors Owners Group (BWROG) has issued a report that provides the basis for this request. This report, NEDO 332977A, dated June 2000 provides justification for relaxation of the surveillance requirement as discussed above. The report demonstrates, through operating experience, a high degree of reliability for EFCV's and the low consequences associated with a failure. PSEG has determined that the experience at Hope Creek is consistent with the finding of the BWROG report.

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in 10CFR50.92. This regulation states that a proposed amendment involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed change has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

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

The proposed requirements affect the surveillance interval for the EFCV's, allowing a reduced number of valves to be tested at each interval. There are no physical plant modifications associated with this change. Industry and Hope Creek experience shows a high degree of reliability for these valves. Because the equipment controlled by the revised Specifications is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident.

Design basis analysis of radiological consequences of an instrument line break outside of primary containment are well within the 10CFR100.11 limits, as defined by acceptance criteria in Standard Review Plan Section 15.7.4. The results of the events remain unchanged from the original design basis, which showed that these events do not result in fuel cladding integrity damage or radioactive releases. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

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

The proposed change in requirements allows a reduced number of EFCV's to be tested each operating cycle. No other changes are being requested. Both Industry and Hope Creek experience demonstrates the high degree of reliability associated with these valves. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. This change will not alter any process variables, structures, systems or components as described in the safety analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The Hope Creek UFSAR, in Section 15.6.2 evaluates an instantaneous circumferential break of an instrument line that is connected to primary coolant. The analysis further assumes that the break can not be isolated, is outside of primary containment and is in a location where the break may not be immediately detected. This analysis assumes the reactor is at full load at the start of the event and that as a consequence of the accident, thereactor vessel is tripped, cooled and depressurized over a 5-hour period. Safety margins and analytical conservatisms have been evaluated and are well understood. Substantial conservatism is retained to ensure that the analysis adequately bounds all postulated event scenarios. The current margin of safety is retained.

The proposed requirements continue to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries as well as control room doses are at or below the corresponding licensing limit. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

CONCLUSION

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

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INSERVICE TESTING PROGRAM - RELIEF REQUEST V-005

COMPONENTS: Excess Flow Check Valves - See Below for Component Numbers

FUNCTION:

These valves are instrument line excess flow check valves installed to meet primary containment isolation requirements. The valves are normally open to provide safety-related instrumentation indications and control functions. The valves automatically close on rising flow (2-3 gpm) to isolate the instrument line from the reactor vessel in the event of an instrument line break.

CATEGORY: C

CLASS: 1

TEST REQUIREMENTS:

Category C check valves shall be exercised nominally every 3 months, in accordance with the requirements of OM-10, paragraph 4.3.2.1.

DEFERRED TEST JUSTIFICATION:

Excess flow check valves are installed on instrument lines penetrating containment in accordance with Regulatory Guide 1.11. The lines are sized and/or orificed such that off-site doses will be substantially below 10CFR100 limits in the event of a rupture. Therefore, individual leak rate testing of these valves is not required for conformance with 10CFR50, Appendix J requirements.

Functional testing of valves to verify closure can be accomplished by the process of venting the instrument side of the valve while the process side is under pressure. Such testing is required by Technical Specification 4.6.3.4 at least once per 18 months. Systems design does not include test taps upstream of the Excess Flow Check Valves. For this reason, the EFCV's cannot be isolated and tested using a pressure source other than reactor pressure. Testing on a frequency greater than once per 18 months is not prudent for several reasons. The testing described above requires the removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and also will contribute to an increase in personnel radiation exposure.

Testing on a quarterly basis is deemed impractical since the risk of performing the test quarterly outweighs the benefit achieved with a quarterly test and will also increase personnel exposure.

Testing on a Cold Shutdown frequency is also impractical considering the large number of valves to be tested and the condition that reactor pressure > 500 psig is needed for testing. OMa – Part 10 – Section 4.2.1.2(e) allows test deferrals to refueling outages if it is impractical to test quarterly or during cold shutdowns.

Industry experience, as documented in NEDO-32977-A, indicates that EFCV's have a very low failure rate. A review of the maintenance history for Hope Creek EFCV's has shown that they have been extremely reliable over the life of the plant, showing less than 1% failure rate associated with testing of these valves. Examples of causes for the failures included alarm problems, indication (limit switch adjustments), and bent instrument tubing. Failures resulted in the replacement of only one of the valves. This review of the surveillance test history shows no evidence of time based failure mechanisms or common mode failures associated with the excess flow check valves. The Hope Creek test experience is consistent with the findings in the NEDO document. The NEDO document indicates similarity that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCV's at Hope Creek, consistent with the industry, have exhibited a high degree of reliability, availability, and provide and acceptable level of quality and safety.

Therefore, PSEG Nuclear LLC requests relief pursuant to 10CFR50.55a(a)(3)(i) to test excess flow check valves at the frequency specified in the Hope Creek Technical Specifications Surveillance Requirements (SR) 4.6.3.4. As discussed in the Technical Specification Bases for this SR, this test provides assurance that each valve actuates to check flow on a simulated instrument line break.

ALTERNATE TESTING:

Functional testing with verification that flow is checked will be performed per Technical Specification 4.6.3.4.

The EFCV's have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification 4.6.3.4. After the close position test, the valves will be reset, and the remote open position indication will be verified. Inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design. Hope Creek verifies that EFCV's indicate open in the control room at a frequency greater than once every 2 years.

COMPONENTS:

1ABXV-3666A	1BBXV-3732F	1BBXV-3803A
1ABXV-3666B	1BBXV-3732G	1BBXV-3803B
1ABXV-3666C	1BBXV-3732H	1BBXV-3803C
1ABXV-3666D	1BBXV-3732J	1BBXV-3803D
1ABXV-3667A	1BBXV-3732K	1BBXV-3804A
1ABXV-3667B	1BBXV-3732L	1BBXV-3804B
1ABXV-3667C	1BBXV-3732M	1BBXV-3804C
1ABXV-3667D	1BBXV-3732N	1BBXV-3804D
1ABXV-3668A	1BBXV-3732P	1BBXV-3820
1ABXV-3668B	1BBXV-3732R	1BBXV-3821
1ABXV-3668C	1BBXV-3732S	1BBXV-3826
1ABXV-3668D	1BBXV-3732T	1BBXV-3827
1ABXV-3669A	1BBXV-3732U	1BCXV-4411A
1ABXV-3669B	1BBXV-3732V	1BCXV-4411B
1ABXV-3669C	1BBXV-3732W	1BCXV-4411C
1ABXV-3669D	1BBXV-3734A	1BCXV-4411D
1BBXV-3621	1BBXV-3734B	1BCXV-4429A
1BBXV-3649	1BBXV-3734C	1BCXV-4429B
1BBXV-3725	1BBXV-3734D	1BCXV-4429C
1BBXV-3726A	1BBXV-3737A	1BCXV-4429D
1BBXV-3726B	1BBXV-3737B	1BEXV-F018A
1BBXV-3727A	1BBXV-3738A	1BEXV-F018B
1BBXV-3727B	1BBXV-3738B	1BGXV-3882
1BBXV-3728A	1BBXV-3783	1BGXV-3884A
1BBXV-3728B	1BBXV-3785	1BGXV-3884B
1BBXV-3729A	1BBXV-3787	1BGXV-3884C
1BBXV-3729B	1BBXV-3789	1BGXV-3884D
1BBXV-3730A	1BBXV-3801A	1FCXV-4150A
1BBXV-3730B	1BBXV-3801B	1FCXV-4150B
1BBXV-3731A	1BBXV-3801C	1FCXV-4150C
1BBXV-3731B	1BBXV-3801D	1FCXV-4150D
1BBXV-3732A	1BBXV-3802A	1FDXV-4800A
1BBXV-3732B	1BBXV-3802B	1FDXV-4800B
1BBXV-3732C	1BBXV-3802C	1FDXV-4800C
1BBXV-3732D	1BBXV-3802D	1FDXV-4800D
1BBXV-3732E		

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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
4.6.3.4	3/4 6-18

Replace Current Paragraph with Following:

Insert A

At least once per 18 months, verify that a representative sample of reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 actuates to the isolation position on a simulated instrument line break signal.

SURVEILLANCE REQUIREMENTS

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

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

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

INSERT A

4.6.3.4 ~~Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.~~

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

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

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

*Exemption to Appendix J of 10 CFR Part 50.

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TECHNICAL SPECIFICATION BASES PAGES WITH CHANGES

Bases (for info only)

Technical Specification	Page
3/4.6.3	B 3/4 6-5
3/4.6.5 (material relocated)	B 3/4 6-6

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Add new paragraph:

Insert B

Surveillance 4.6.3.4 requires demonstration that a representative sample of reactor instrumentation line excess flow check valves are tested to demonstrate that the valve actuates to check flow on a simulated instrument line break. This surveillance requirement provides assurance that the instrument line EFCV's will perform so that the predicted radiological consequences will not be exceeded during a postulated instrument line break event as evaluated in the UFSAR. The 18-month frequency is based on the need to perform this surveillance under the conditions that apply immediately prior to and during the plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. The representative sample consists of an approximately equal number of EFCV's, such that each EFCV is tested at least once every ten years (nominal). In addition, the EFCV's in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. The nominal 10 year interval is based on performance testing as discussed in NEDO 32977-A, "Excess Check Valve Testing Relaxation." Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

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.

INSERT B

3/4.6.4 VACUUM RELIEF

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.