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NLR-N94154
LCR 94-22

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

Gentlemen:

LICENSE AMENDMENT APPLICATION
DELETION OF CERTAIN VALVES ON LINES SERVING
CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM
TECHNICAL SPECIFICATION TABLE 3.6.3-1
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

This letter submits an application for amendment to Appendix A of Facility Operating License NPF-57 for the Hope Creek Generating Station (HCGS), and is being filed in accordance with 10CFR50.90. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed amendment involves the reclassification of certain Containment Isolation Valves (CIVs) on lines that serve closed, Safety Class 2 Engineered Safety Feature (ESF) systems outside containment. The proposed amendment deletes these reclassified valves from the list of CIVs provided in Technical Specification (TS) Table 3.6.3-1. This request reflects aspects of the License Amendment issued by the NRC on April 22, 1994 for Detroit Edison Company's Fermi-2 Nuclear Plant (Docket No. 50-341).

PSE&G considers this submittal a CBLA since the removal of certain valves from TS Table 3.6.3-1 eliminates unnecessary 10CFR50 Appendix J testing. We have estimated the cost savings as $520,800/yr at the HCGS. Savings over the life of the plant are $16,665,600. The deletion of the subject valves from TS Table 3.6.3-1 is expected to reduce exposures by 1.7 R/outage, with resultant ALARA dose improvements.

This proposed change has been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c) and it has been determined that this request involves no significant hazards considerations.

A description of the requested amendment, supporting information and analyses for the change, and the basis for a no significant hazards consideration determination are provided in Attachment 1. The Technical Specification pages affected by the proposed change are marked in Attachment 2.

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Upon NRC approval of this proposed change, PSE&G requests that the amendment be made effective on the date of issuance, but implemented within sixty (60) days to provide sufficient time for associated administrative activities in support of Refueling Outage 6, which is scheduled to begin in September 1995.

Should you have any questions regarding this request, we will be pleased to discuss them with you.

Sincerely,

Stanley LaBruna
Vice President -
Nuclear Engineering

Affidavit
Attachments (2)

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REF: NLR-N94154
LCR 94-22

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

Stanley LaBruna, being duly sworn according to law deposes and says:

I am Vice President - Nuclear Engineering of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning the Hope Creek Generating Station, are true to the best of my knowledge, information and belief.

S. LaBruna

Subscribed and Sworn to before me
this 31 day of October, 1994

Elizabeth J. Kidd
Notary Public of New Jersey

My Commission expires on 4/25/95

ELIZABETH J. KIDD
Notary Public of New Jersey
My Commission Expires April 25, 1995

ATTACHMENT 1
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

LICENSE AMENDMENT APPLICATION
DELETION OF CERTAIN VALVES ON LINES SERVING
CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM
TECHNICAL SPECIFICATION TABLE 3.6.3-1
HOPE CREEK GENERATING STATION — NLR-N94154
FACILITY OPERATING LICENSE NPF-57 — LCR 94-22
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I. DESCRIPTION OF THE PROPOSED CHANGES

As indicated on the marked-up pages in Attachment 2, PSE&G requests that certain valves be reclassified and deleted from Technical Specification (TS) Table 3.6.3-1 "Primary Containment Isolation Valves." The subject valves are located on lines which serve closed, Safety Class 2 Engineered Safety Feature (ESF) systems. In addition to the subject valves, each of these lines is provided with an automatic Containment Isolation Valve (CIV) outside Containment which remains subject to TS Table 3.6.3-1.

The ESF piping systems constitute closed systems outside Primary Containment (Containment). The use of closed, extended Containment boundary system piping outside Containment, in conjunction with the use of a single, automatic CIV outside Containment, provides a dual isolation barrier which satisfies the "other defined basis" requirements of 10CFR50 Appendix A General Design Criteria 55 and 56.

The proposed change does not affect the design, functions, and operation of the subject valves. Under this change, the subject valves will remain fully operational and capable of performing all of their required system and safety-related functions. Those valves, which are currently identified as Reactor Coolant System Pressure Isolation Valves, will continue to be subject to Technical Specification 3/4.4.3.2. The valves will continue to be inspected and tested per the applicable requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME XI) in accordance with Technical Specification 4.0.5 under this change.

II. REASON FOR THE CHANGES

The subject license amendment application is a Cost Beneficial Licensing Action since the removal of certain valves from TS Table 3.6.3-1 eliminates unnecessary 10CFR50 Appendix J leak rate testing. Such testing is not necessary to ensure that post-accident radiological releases from the Containment are minimized consistent with the existing accident analyses. The elimination of the 10CFR50 Appendix J leak rate testing for the subject valves will also reduce occupational radiation exposures while continuing to assure Containment integrity.

Note that this amendment request is not an exemption to 10CFR50 Appendix J requirements. Following a review of the isolation valve configurations used in ESF systems, PSE&G has determined that the HCGS design satisfies the "other defined basis" containment isolation provisions of 10CFR50 Appendix A General Design Criteria (GDC) 55 and 56 as described in Section III.

III. JUSTIFICATION FOR THE CHANGES

The Residual Heat Removal (RHR), Core Spray (CS) and High Pressure Coolant Injection (HPCI) Systems are among the Engineered Safety Feature (ESF) systems included in the HCGS design. The RHR System includes various normal and post-accident operational modes including Shutdown Cooling, Low Pressure Coolant Injection, Head Spray, and Containment Spray. The CS System is used post-accident. In addition, a HPCI line discharges into the Reactor Vessel via one of the two CS lines downstream of the outboard CIV on that CS line. The RHR and the CS Systems deliver cooling water to the Reactor following postulated accidents when the Reactor is at low pressure conditions. The RHR System also delivers cooling water to the Containment to reduce pressure post-accident. The HPCI System is used to deliver cooling water to the Reactor following postulated accidents when the Reactor is at high pressure.

In performing these functions, the RHR, CS and HPCI Systems penetrate the Primary Containment Pressure Boundary and the Reactor Coolant Pressure Boundary (RCPB), as necessary. In order to allow the normal and emergency passage of water through the Containment, while preserving the ability to prevent or limit the escape of fission products following a postulated accident, Containment isolation provisions are included in the system designs.

The RHR, CS and HPCI Systems are designed to satisfy the appropriate Containment isolation requirements including those contained in 10CFR50 Appendix A General Design Criteria (GDC) 55 and 56. The current licensing basis satisfies these GDCs through the use of redundant CIVs.

A CIV is a valve which could isolate a potential fission product release pathway to the environment following a postulated accident and, consequently, its allowable leakage should be minimized. Among the CIVs included in TS Table 3.6.3-1 are those associated with the subject RHR, CS and HPCI lines. Currently, these CIVs are subject to 10CFR50 Appendix J, Type A and C testing in accordance with Technical Specifications 3/4.6.1.1, 3/4.6.1.2 and 3/4.6.3.

In addition, some of the valves included in this amendment request are also Reactor Coolant System Pressure Isolation Valves and therefore subject to Technical Specification 3/4.4.3.2. The

valves are also subject to the inservice inspection and testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code (ASME XI) required by Technical Specification 4.0.5.

As described in subsequent paragraphs, the Safety Class 2, seismic Category I RHR, CS and HPCI Systems are Emergency Core Cooling Systems (ECCS). Portions of the systems that are part of the Reactor Coolant Pressure Boundary are Safety Class 1, seismic Category I. The systems may be placed in service following a postulated accident and they would circulate radioactive water from the Suppression Pool. Accordingly, the RHR, CS and HPCI pipes outside Containment are designed as extended Containment boundaries.

The RHR, CS and HPCI Systems were also designed using redundant CIVs, usually configured with one CIV inside, and the other CIV outside, the Containment. These configurations satisfy the applicable GDC 55 and 56 requirements. However, GDC 55 and 56 also allow the use of other, acceptable containment isolation provisions if they are provided on some "other defined basis."

The Containment isolation provisions for the subject lines have been reviewed against the guidance provided in NUREG-0800 "Standard Review Plan" Section 6.2.4 "Containment Isolation System" Paragraph II.6.e and Regulatory Guide (RG) 1.141 Rev. 0, April 1976 "Containment Isolation Provisions for Fluid Systems" with regard to extended Containment boundaries. Section 3.6.4 of the ANS-56.2/ANSI N271-1976 "Containment Isolation Provisions for Fluid Systems" Standard referenced in this RG establishes that a single valve and closed system, both outside Containment, is an acceptable alternative configuration for complying with the GDC requirements. This alternative configuration is the "other defined basis" which allows the reclassification and deletion of the interior CIVs, or for the outermost CIV in the case of the RHR Containment Spray, from TS Table 3.6.3-1 "Primary Containment Isolation Valves." Note that the HCGS also complies with Section 3.6.4 of ANSI/ANS-56.2-1984 "Containment Isolation Provisions for Fluid Systems After a LOCA", the latest version of the N271 Standard.

The specific Containment isolation boundary considerations, which support the removal of the subject valves from the list of CIVs, are summarized in Table 1. The premise behind PSE&G's proposed amendment is that the use of one automatic CIV outside Containment, in conjunction with the use of water-filled, Safety Class 2, seismic Category I (i.e., closed system/extended Containment boundary) piping outside Containment, constitutes a valid, redundant, isolation barrier. The proposed amendment reclassifies the subject valves and deletes them from TS Table 3.6.3-1. This eliminates unnecessary 10CFR50 Appendix J leak rate testing for the valves specifically identified in Table 1. This will result in a reduction in occupational radiation exposures while continuing to assure Containment integrity.

The deletion of the subject valves from TS Table 3.6.3-1 will not impact the existing accident analysis. The proposed change does not affect the design, functions and operation of these valves. Other applicable requirements, pertaining either to the valves or the systems they serve which are imposed by the Technical Specifications, are not affected by this amendment request.

The following paragraphs discuss defense-in-depth considerations which justify the deletion of the subject valves from TS Table 3.6.3-1:

Valve Functions

Although the valves subject to this amendment application will no longer be considered CIVs, the proposed change does not affect the design, functions, and operation of these valves. The valves will remain fully operational and capable of performing all of their required system and safety-related functions.

The valves are designed to be opened post-accident, as necessary, in order to properly align the RHR, CS and HPCI Systems for the performance of their required safety functions. Since the RHR, CS and HPCI Systems are designed to circulate radioactive water from the Suppression Pool following postulated accidents, post-accident leakage past these valves has been considered in the design and is not a radiological concern from a Containment isolation perspective. (Refer to the following "Containment Leakage Barriers" and "Other Considerations" discussions for additional clarification).

The leak tight integrity of certain of these valves as high-low pressure interface boundaries will continue to be demonstrated by their performance during normal operation. These valves remain subject to TS 3/4.4.3.2. In addition, all of the valves included in this amendment application remain subject to ASME XI inservice testing and inspection in accordance with TS 4.0.5.

Closed System Piping

The RHR, CS and HPCI Systems are ESF systems that circulate water from the Suppression Pool following postulated accidents. The RHR, CS and HPCI Systems are designed as closed systems outside the Containment and are physically located in the Reactor Building. The adequacy of these closed piping systems outside Containment as extended Containment boundaries is evaluated in HCGS UFSAR Section 6.2.4.3.5. The water-filled, Safety Class 2, seismic Category I piping, in conjunction with the automatic CIVs outside Containment, precludes any direct communication between the post-accident Containment atmosphere and the Reactor Building under the various post-accident scenarios, including postulated single failures.

HCGS UFSAR Section 6.3.2.2.6 states that the ECCS Discharge Line Fill Network is designed to maintain the ECCS pump discharge lines in a filled condition. The RHR, CS and HPCI piping outside Containment are served by this fill network. Consequently, these lines are maintained full of water either due to the operation of the fill network or as a result of the RHR, CS and HPCI Systems being placed in service. Since the piping is maintained full of water during normal and post-accident modes of operation, a barrier against post-accident, gaseous, Containment leakage is provided.

Containment Leakage Barriers

The use of automatic CIVs outside Containment, in conjunction with the use of water-filled, closed system/extended Containment boundary piping outside Containment, ensures that leakage pathways from the Containment following a postulated accident do not exist. These redundant isolation barriers are located outside Containment in the Reactor Building. Together, they provide protection against postulated single active and passive failures and satisfy the requirements of GDC 55 and 56.

The CIVs for these systems, which will continue to be included in TS Table 3.6.3-1, and the associated closed piping systems are all located outside the Containment in the radiologically-controlled Reactor Building. This building is served by the Filtration, Recirculation, and Ventilation System (FRVS). The FRVS is designed to minimize post-accident radiological releases to the environment.

Any post-accident effluent releases due to seat leakage of the subject valves would be confined within the system piping. Releases associated with gland leakage would be within the Drywell for the subject valves located inside the Containment. Gland leakage for the valves located outside the Containment is confined within the Reactor Building where it would be filtered and monitored by the FRVS prior to release from the Plant. The radiological consequences associated with these releases are within the existing plant licensing bases.

Also note that any post-accident leakage past the valves would usually be expected to involve liquid, not gaseous, releases of the Containment atmosphere. The water within the RHR, CS and HPCI piping helps to minimize post-accident fission product releases to the Reactor Building. However, Type C testing using gas will continue for the CIVs as specified in TS Table 3.6.3-1. This assures that even if gaseous effluents are present, the CIVs are tested in an appropriate manner to minimize the potential for leakage. Therefore, a significant radiological consequence would not occur.

Other Considerations

The RHR, CS and HPCI Systems are included in the Leakage Reduction Program described in Technical Specification 6.8.4.a. This program is implemented to reduce leakage from systems outside the Containment that could contain highly radioactive fluids following an accident. Program elements include design features to minimize leakage; instrumentation to detect gross leakage within the Reactor Building; visual examination during system operation; periodic leakage tests; a vigorous corrective action program to correct leakage problems once they have been identified; and preventive maintenance activities. These elements will detect and correct degradation of the pressure boundaries of the subject systems, thereby reducing post-accident releases and resultant dose consequences in the Reactor Building. This would also reduce the amount of radioactivity processed and released by the FRVS.

With the exception of the valves serving the RHR Containment Spray System (Penetrations 24A & B), the subject valves are all located within the Containment and are Reactor Coolant System Pressure Isolation Valves (PIVs). Operational Leakage Control per TS 3/4.4.3.2 apply to those subject valves, which are also PIVs.

Although the subject valves will not be tested under PSE&G's 10CFR50 Appendix J testing program, the overall integrated leak-tight integrity of the Primary Reactor Containment will be confirmed during 10CFR50 Appendix J, Type A testing.

The ability of the valves to perform required functions will be demonstrated during normal plant operation and via surveillance testing in accordance with the applicable Technical Specification requirements. Inservice inspection and testing will continue to be conducted in accordance with the appropriate requirements of ASME XI.

IV. SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

PSE&G has, pursuant to 10CFR50.92, reviewed the proposed amendment to determine whether our request involves a significant hazards consideration. We have determined that the operation of the Hope Creek Generating Station (HCGS) in accordance with the proposed changes associated with reclassifying certain of the redundant, Containment Isolation Valves (CIVs) on lines that serve water-filled, Safety Class 2, seismic Category I closed (extended Containment boundary) ESF Systems piping outside Containment and deleting these valves from Technical Specification Table 3.6.3-1 "Primary Containment Isolation Valves":

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The valves, which will be deleted from TS Table 3.6.3-1, serve water-filled, Safety Class 2, seismic Category I ESF system lines. Safety Class 2, seismic Category I ESF system lines utilize other, automatic CIVs outside the Containment in addition to the valves subject to this amendment application. These other CIVs will continue to be listed in TS Table 3.6.3-1 and subject to the requirements specified in the Technical Specifications. In addition, the piping serving these ESF systems constitute closed systems (i.e., extended Containment boundaries) outside Containment, as described in HCGS UFSAR Section 6.2.4.3.5. The dual barriers provided by the CIVs and closed system piping outside the Containment satisfy the "other defined basis" requirements of GDC 55 and 56, as clarified in RG 1.141 and the ANSI/ANS Standard referenced therein. These dual barriers prevent post-accident Containment leakage and provide protection against postulated single active and passive failures. Therefore, the post-accident integrity of the Containment can be assured following the removal of the subject valves from TS Table 3.6.3-1.

   The elimination of the subject valves from the list of CIVs will not increase the potential for containment leakage due to the presence of the other CIVs and use of closed system piping. The proposed change does not affect the design, functions, and operation of these valves. The valves will remain fully operational and capable of performing all of their required system and safety-related functions. Other applicable requirements, pertaining either to the valves or the systems they serve which are imposed by Technical Specifications, are not affected by this amendment application.

   Certain valves, which are Reactor Coolant System Pressure Isolation Valves, remain subject to Technical Specification 3/4.4.3.2. In addition, all of the valves included in this amendment application will continue to be inspected and tested per the applicable requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME XI) in accordance with Technical Specification 4.0.5.

   The changes associated with this amendment application will not affect the overall leak-tight integrity of the Primary Reactor Containment, which will be demonstrated during 10CFR50 Appendix J Type A Integrated Leak Rate Testing as described in the Hope Creek Generating Station Updated Final Safety Analysis Report (HCGS UFSAR) Section 6.2.6.1.

Radiological releases and their consequences due to leakage of the subject valves will be within the existing plant licensing basis.

The proposed change does not change the physical plant or the manner in which it is operated. The change eliminates unnecessary Appendix J testing requirements for certain valves while retaining those tests needed to assure required Containment isolation capabilities. The new bases for satisfying GDC 55 and 56 provide equivalent levels of protection against offsite radiation releases.

2. Will not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposal does not involve any hardware or logic changes, nor does it alter the way in which any plant systems operate. Post-accident Containment isolation features, boundaries and system interfaces are not affected by the changes. The change eliminates unnecessary Appendix J tests for certain valves while retaining those tests needed to assure that the Containment isolation features perform as required. Therefore, there are no new possibilities or types of accidents considered.

3. Will not involve a significant reduction in a margin of safety.

The proposed elimination of certain valves from the list of isolation valves contained in Technical Specification Table 3.6.3-1 will not adversely affect the margins of safety associated with the plant's licensing bases.

The use of other CIVs outside Containment, which are subject to Type C testing with gas in accordance with the Technical Specifications, in conjunction with the closed system/extended Containment boundary piping, provides a redundant barrier against Containment leakage and provides protection against postulated single active and passive failures. The subject valves will be tested in accordance with the ASME B&PV Code Section XI - Division 1, Article IWV-3000, as required. 10CFR50 Appendix J, Type A testing will ensure that the overall Containment leakage rate is consistent with the plant's licensing bases. Existing Containment isolation features, boundaries and system interfaces are not affected by the change.

Since the valves and the systems they serve are located either in the Drywell or in the Reactor Building, any leakage (e.g., packing gland leakage) which escapes the confines of the closed system piping, will be contained within the Containment or in the Reactor Building. These areas are radiologically controlled and monitored. Any

releases to the Reactor Building are processed by the Filtration, Recirculation, and Ventilation System. This assures that all radioactive releases to the environment are within the existing plant licensing bases.

The elimination of unnecessary Appendix J testing for certain valves will not affect the existing radiological release evaluations currently described in the HCGS-UFSAR.

The proposed Amendment will not affect the functional capability of any plant safety-related structures, systems or components.

In addition, the implementation of the proposed Amendment will result in a reduction of radiological exposure to plant personnel. Therefore, the proposed revision will not reduce a margin of safety.

V. CONCLUSION

Based on the previous discussion, PSE&G has concluded that the proposed change to the Technical Specifications does not involve a significant hazards consideration insofar as the change: (i) does not involve a significant increase in the probability or consequences of an accident previously evaluated, (ii) does not create the possibility of a new or different kind of accident from any accident previously evaluated, and (iii) does not involve a significant reduction in a margin of safety.

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-3, RHR Shutdown Cooling Suction Line

VALVES: BC-V071 (HV-F009) - RHR Shutdown Cooling Suction from RPV
BC-PSV-4425 - RHR Shutdown Cooling Suction Line Thermal Relief Valve.

Function: These Safety Class 1, seismic Category I valves are located in the Drywell in the piping which supplies water from one of the reactor recirculation lines to the RHR pump suction nozzles. They are part of the RCPB.

These valves are normally closed. Motor operated valve BC-V071 is opened remote-manually from the Control Room in order to remove decay and sensible heat from the Reactor to allow refueling. This valve may also be opened post-accident remote-manually from the Control Room as necessary. Pressure relief valve BC-PSV-4425 may open in response to elevated pressure in the penetration piping due to heating of the water trapped between valve BC-V071 and the closed CIV located outside the Containment (BC-V164 (HV-F008)). During a postulated accident, valve BC-V071 closes automatically in response to low reactor water level. During normal operation, the valve is interlocked to close on high reactor pressure.

As indicated above, the RHR shutdown cooling suction line also utilizes a motor-operated CIV outside Containment. This CIV, BC-V164, is not affected by the amendment application and will continue to be subject to TS Table 3.6.3-1. The Safety Class 2, seismic Category I RHR piping outside Containment is an extended Containment boundary. Although the piping outboard of the penetration (i.e., between valves BC-V164 (HV-F008) and BC-V104 (HV-F006A) / BC-V007 (HV-F006B)) is not served by the Fill Network and may not normally be full of water, the RHR piping outside Containment, which is downstream of this piping, is served by the Fill Network and is full of water during normal and post-accident plant operations.

Comments*: 1, 2, 3, 4, 5, 7, 8, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.1.3; Fig. 5.4-13 sh. 1 of 2, Fig. 6.2-47 sh. 1 of 3 and Fig. 6.2-27 sh. 3 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BC-V071 - ASME XI Category A.
BC-PSV-4425 - ASME XI Categories A and C.

(Note that BC-PSV-4425 is subject to 10CFR50 Appendix J Type A Containment Integrated Leak Rate Testing).

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-4A, RHR Shutdown Cooling Return Line
P-4B, RHR Shutdown Cooling Return Line

Valves: BC-V014 (HV-F050B) - RHR Shutdown Cooling Return Line Testable Check Valve (P-4A)
BC-V111 (HV-F050A) - RHR Shutdown Cooling Return Line Testable Check Valve (P-4B)
BC-V118 (HV-F122B) - Bypass Line Globe Valve on RHR Shutdown Cooling Return Lines (P-4A)
BC-V117 (HV-F122A) - Bypass Line Globe Valve on RHR Shutdown Cooling Return Lines (P-4B)

Function: These Safety Class 1, seismic Category I valves are located in the Drywell in the piping which supplies water from the RHR pumps to the reactor recirculation lines. They are part of the RCPB.

Check valves BC-V014 and BC-V111 are normally closed but are open in response to system flow when the RHR System is in the shutdown cooling mode of operation. The operators for these valves are only utilized to exercise the valve discs during testing. These valves may also be open post-accident in response to system flow.

Air-operated, fail-closed bypass globe valves BC-V118 and BC-V117 are provided to equalize the pressure across the discs of check valves BC-V014 and BC-V111, respectively, during testing. Otherwise, these valves are closed.

The RHR System also utilizes a motor-operated CIV outside Containment on each of the two lines. These CIVs, BC-V013 (HV-F015B) and BC-V110 (HV-F015A), are not affected by the amendment application and will continue to be subject to TS Table 3.6.3-1. The Safety Class 2, seismic Category I RHR piping outside Containment is an extended Containment boundary. The RHR piping outside Containment is filled with water during normal and post-accident plant operations.

Comments*: 1, 2, 3, 4, 5, 7, 8, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.1.4; Fig. 5.4-13 sh. 1 and 2 of 2, Fig. 6.2-47 sh. 1 and 2 of 3, Fig. 6.2-27 sh. 4 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BC-V118 and BC-V117 - ASME XI Category A.
BC-V014 and BC-V111 - ASME XI Categories A and C.

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-5A, Core Spray Injection Line
P-5B, Core Spray Injection Line

Valves: BE-V002 (HV-F006B) - Core Spray Injection Line Testable Check Valve (P-5A)
BE-V006 (HV-F006A) - Core Spray Injection Line Testable Check Valve (P-5B)
BE-V072 (HV-F039B) - Bypass Line Globe Valve for Testable Check Valve BE-V002 (P-5A)
BE-V071 (HV-F039A) - Bypass Line Globe Valve for Testable Check Valve BE-V006 (P-5B)

Function: These Safety Class 1, seismic Category I valves are located in the Drywell in the piping which supplies water from the Core Spray (CS) pumps to the Reactor. Note that Penetration P-5B may also be used to supply water from the HPCI pump to the Reactor via the CS lines. The valves are part of the RCPB.

Check valves BE-V006B and BE-V006A are normally closed but are open in response to system flow when the CS and/or HPCI pumps are placed in operation following an accident. The operators for these valves are only utilized to exercise the valve discs during testing.

Air-operated, fail-closed bypass globe valves BE-V072 and BE-V071 are provided to equalize the pressure across the discs of check valves BE-V002 and BE-V006, respectively, during testing. Otherwise, these valves are closed.

The CS lines also utilize motor-operated CIVs outside Containment. CIV BE-V003 (HV-F005B) serves penetration P-5A, while CIVs BE-V007 (HV-F005A) and BJ-V001 (HV-F006) serve P-5B. These CIVs are not affected by the amendment application and will continue to be subject to TS Table 3.6.3-1. The Safety Class 2, seismic Category I CS and HPCI piping outside Containment are extended Containment boundaries. The CS and HPCI piping outside Containment is filled with water during normal and post-accident plant operations.

Comments*: 1, 2, 3, 4, 5, 7, 8, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.1.5; Fig. 6.3-1 sh. 1 of 1, Fig. 6.3-7 sh. 1 of 1, Fig. 6.2-46 sh. 1 of 2, Fig. 6.2-47 sh. 3 of 3, Fig. 6.2-27 sh. 5 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BE-V072 and BE-V071 - ASME XI Category A.
BE-V002 and BE-V006 - ASME XI Categories A and C.

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-6A, RHR LPCI Injection Line
P-6B, RHR LPCI Injection Line
P-6C, RHR LPCI Injection Line
P-6D, RHR LPCI Injection Line

Valves:
- BC-V005 (HV-F041D) - RHR LPCI Injection Line Testable Check Valve (P-6A)
- BC-V017 (HV-F041B) - RHR LPCI Injection Line Testable Check Valve (P-6B)
- BC-V114 (HV-F041A) - RHR LPCI Injection Line Testable Check Valve (P-6C)
- BC-V102 (HV-F041C) - RHR LPCI Injection Line Testable Check Valve (P-6D)
- BC-V122 (HV-F146D) - Bypass Line Globe Valve for Testable Check Valve BC-V005 (P-6A)
- BC-V120 (HV-F146B) - Bypass Line Globe Valve for Testable Check Valve BC-V017 (P-6B)
- BC-V119 (HV-F146A) - Bypass Line Globe Valve for Testable Check Valve BC-V114 (P-6C)
- BC-V121 (HV-F146C) - Bypass Line Globe Valve for Testable Check Valve BC-V102 (P-6D)

Function: These Safety Class 1, seismic Category I valves are located in the Drywell in the piping which supplies water from the RHR pumps to the Reactor. They are part of the RCPB.

Check valves BC-V005, BC-V017, BC-V114, and BC-V102 are normally closed but are open in response to system flow when the RHR System is utilized in the LPCI mode of operation following an accident. The operators for these valves are only utilized to exercise the valve discs during testing.

Air-operated, fail-closed bypass globe valves BC-V122, BC-V120, BC-V119 and BC-V121 are provided to equalize the pressure across the discs of check valves BC-V005, BC-V017, BC-V114, and BC-V102, respectively, during testing. Otherwise, these valves are closed.

The LPCI utilizes a motor-operated CIV outside Containment on each of the four lines. These CIVs, BE-V004 (HV-F017D), BE-V016 (HV-F017B), BE-V113 (HV-F017A), and BE-V101 (HV-F017C), are not affected by the amendment application and will continue to be subject to TS Table 3.6.3-1. The Safety Class 2, seismic Category I RHR piping outside Containment is an extended Containment boundary. The RHR piping outside Containment is filled with water during normal and post-accident plant operations.

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1
(CONTINUED)

Comments*: 1, 2, 3, 4, 5, 7, 8, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.1.5; Fig. 5.4-13 sh. 1 and 2 of 2, Fig. 6.2-47 sh. 1 and 2 of 3, Fig. 6.2-27 sh. 5 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BC-V122, BC-V120, BC-V119, and BC-V121 - ASME XI Category A.
BC-V005, BC-V017, BC-V119 and BC-V102 - ASME XI Categories A and C.

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-10, RHR Head Spray Line

Valves: BC-V021 (HV-F022) - RHR Head Spray Isolation Gate Valve

Function: This Safety Class 1, seismic Category I valve is located in the Drywell in the piping which supplies water from the RHR pumps to the Reactor. The valve is part of the RCPB. The RHR Head Spray is not an ESF function as stated in UFSAR Table 6.2-16, however the RHR System which supplies water for this function is an ESF System.

Motor-operated gate valve BC-V021 is normally closed but may be opened remote-manually from the Control Room when the RHR System is utilized in the shutdown cooling mode of operation in order to limit thermal stress by removing heat from the RPV walls. During a postulated accident, the valve will automatically close in response to low reactor water level. During normal operation, the valve is also interlocked to close on high reactor pressure.

The RHR Head Spray utilizes a motor-operated CIV outside Containment. This CIV, BC-V020 (HV-F023), is not affected by this amendment application and will continue to be subject to TS Table 3.6.3-1. The Safety Class 2, seismic Category I RHR piping outside Containment is an extended Containment boundary. The RHR piping outside Containment is filled with water during normal and post-accident plant operations.

Comments*: 1, 2, 3, 4, 5, 8, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.1.8; Fig. 5.4-13 sh. 1 of 2, Fig. 6.2-47 sh. 1 of 3, Fig. 6.2-27 sh. 8 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BC-V021 - ASME XI Category A.

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

Penetrations: P-24A, RHR Containment Spray
P-24B, RHR Containment Spray

Valves: BC-V018 (HV-F016B) - RHR Containment Spray Supply Line Gate Valve (P-24A)
BC-V115 (HV-F016A) - RHR Containment Spray Supply Line Gate Valve (P-24B)

Function: These Safety Class 2, seismic Category I valves are located outside the Drywell on each of the two Containment Spray supply headers. Each valve is located upstream of the single motor-operated CIV serving each of the lines.

The Containment Spray lines are used to control internal Containment pressure post-accident by supplying water from the RHR pumps to the Containment Spray nozzles located inside the Drywell.

Gate valves BC-V018 and BC-V115 are normally closed. They are interlocked with their corresponding CIVs (i.e., BC-V019 (HV-F021B) and BC-V116 (HV-F021A), respectively) so that only one of the two valves serving each line may be opened at a time during system testing. The valves may only be opened post-accident remote-manually from the Control Room after a high Primary Containment pressure signal is received. The motor-operated valves are provided with Class IE power. The valves do not receive a Containment Isolation Signal and are designed to fail "as is". The valves are closed remote-manually by the operators in the Control Room following termination of the Containment Spray function. CIVs BC-V019 and BC-V116 are not affected by the amendment application and will continue to be subject to TS Table 3.6.3-1.

The RHR System utilizes Safety Class 2, seismic Category I piping outside Containment. This piping is an extended Containment boundary. The RHR piping outside Containment is filled with water during normal and post-accident plant operations.

Comments*: 1, 2, 3, 4, 6, 7, 9, and 10.

HCGS-UFSAR References: Section 6.2.4.3.2.2; Fig. 5.4-13 sh. 1 and 2 of 2, Fig. 6.2-47 sh. 1 and 2 of 3, Fig. 6.2-27 sh. 15 of 49; and Tables 6.2-16 and 24.

ASME Test Designation: BC-V018 and BC-V115 - ASME XI Category A.

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* - See Comments Key (Pages 17 and 18)

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

COMMENTS KEY

1. The penetration includes a CIV outside Containment. This CIV is not affected by the amendment application and is subject to all applicable Technical Specifications including those specified in TS Table 3.6.3-1.

2. The penetration serves a closed system outside Containment, which constitutes a second isolation barrier in addition to the CIV located outside the Primary Containment (refer to Note 1 above). The closed system: a) does not communicate with the outside atmosphere; b) is protected against missiles and pipe whip; c) is designed to seismic Category I requirements per Regulatory Guide (RG) 1.29; d) is classified as a minimum as Quality Group B (refer to Note 7 below) per RG 1.26; e) is evaluated as an extended Containment boundary following the postulated single failure of the CIV (Refer to HCGS UFSAR Section 6.2.4.3.5); f) is designed to withstand temperatures and internal pressures equal to those of the Containment; g) is designed to withstand the transients described in UFSAR Chapter 15; h) is protected against overpressure from thermal expansion of contained fluids when isolated, as applicable; and i) is capable of being leak rate tested.

3. The penetration piping, subject valves and CIVs: a) do not communicate with the outside atmosphere; b) are protected against missiles and pipe whip; c) are designed to seismic Category I requirements per Regulatory Guide (RG) 1.29; d) are classified as a minimum as Quality Group B (refer to Note 7 below) per RG 1.26; e) are designed to withstand temperatures and internal pressures equal to those of the Containment; f) are designed to withstand the transients described in UFSAR Chapter 15; g) are protected against overpressure from thermal expansion of contained fluids when isolated, as applicable; and h) are capable of being leak rate tested.

4. The piping, subject valves and CIV(s) in the penetration areas are conservatively designed to preclude breach of pressure integrity consistent with HCGS UFSAR Section 3.6.2.1.1.1.

5. The subject valve(s) is/are located inside Containment. The associated CIV(s) and closed system piping (extended Containment boundaries) are located outside the Containment in the Reactor Building in an area served by the Filtration, Recirculation, and Ventilation System following an accident.

TABLE 1

ISOLATION BOUNDARY CONSIDERATIONS SUPPORTING THE DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1

COMMENTS KEY (CONTINUED)

6. The subject valves, CIV(s) and associated closed system piping (extended Containment boundaries) are located outside the Containment in the Reactor Building in an area served by the Filtration, Recirculation, and Ventilation System following an accident.

7. The subject valves and CIV(s) may be open post-accident (i.e., seat leakage is not relevant in regard to Containment leakage).

8. The subject valve(s) and CIV(s) are Reactor Coolant System Pressure Isolation Valves. These valves, and the associated penetration piping, are Quality Group A per RG 1.26 and subject to Operational Leakage Control per Technical Specification 3/4.4.3.2.

9. Surveillance requirements for inservice inspection and testing of ASME Code Class 1 and 2 components are as specified in Technical Specification 4.0.5.

10. The CIVs, subject valves, and associated piping which are located outside the Containment, are included in the Leakage Reduction Program described in Technical Specification 6.8.4.a.

ATTACHMENT 2
LICENSE AMENDMENT APPLICATION
DELETION OF CERTAIN VALVES ON LINES SERVING CLOSED ESF SYSTEMS
OUTSIDE CONTAINMENT FROM TECHNICAL SPECIFICATION TABLE 3.6.3-1
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

TECHNICAL SPECIFICATION PAGES WITH PEN AND INK CHANGES

The following Technical Specification pages for Facility Operating License No. NPF-57 are affected by this License Amendment Request:

Technical Specification	Pages
Table 3.6.3-1	3/46-21
	3/46-30
	3/46-31
	3/46-32
	3/46-36