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

PSEG
Nuclear LLC

10 CFR 50.55(a)

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

Hope Creek Generating Station
Renewed Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: Request for Additional Information Regarding Relief Requests Associated with the Fourth 10-Year Inservice Test Interval

References:

1. PSEG Letter LR-N15-0250, "Inservice Testing (IST) Program - Fourth Ten-Year Interval," dated December 18, 2015 (ADAMS Accession No. ML15352A127)

2. NRC Letter to PSEG, "Hope Creek Generating Station - Request for Additional Information Regarding Relief Requests GR-01, PR-01, PR-02, VR-01, and VR-02, Associated with the Fourth 10-Year Inservice Test Interval (CAC Nos. MF7200, MF7201, MF7202, MF7203, and MF7204)," dated May 5, 2016 (ADAMS Accession No. ML16089A079)

In the Reference 1 letter, PSEG Nuclear LLC (PSEG) submitted relief requests to the U.S. Nuclear Regulatory Commission (NRC) for the Hope Creek Generating Station. The requests proposed alternatives to the requirements of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants, 2012 Edition with no Addenda, for the fourth 10-year inservice testing (IST) program interval.

In the Reference 2 letter, the NRC staff requested additional information for two of the proposed alternatives. PSEG's responses are provided in Attachment 1.

The proposed alternatives in Reference 1 were to the requirements of the ASME OM - 2012 Edition, based on PSEG's understanding of the timetable to complete the current rulemaking to incorporate the 2012 Edition by reference into 10 CFR 50.55a. However, due to uncertainty as to when the final rule will be issued, PSEG is electing to change the applicable code edition to the 2004 Edition with 2006 Addenda, which is the latest edition and addenda of the OM Code currently incorporated by reference in 10 CFR 50.55a(a)(1)(iv). Resulting changes in the applicable code requirements for each proposed alternative are described in Attachment 2.

As noted in Attachment 1, PSEG will exclude three-stage Target Rock main steam pressure relief valves from the scope of 10 CFR 50.55a Request VR-02.

There are no regulatory commitments contained in this letter. If you have any questions or require additional information, please contact Mr. Lee Marabella at (856) 339-1208.

Sincerely,

Paul R. Duke, Jr.
Manager, Licensing

Attachments

1. PSEG Response to Request for Additional Information Regarding Relief Requests VR-01 and VR-02, Associated With the Fourth 10-Year Inservice Test Interval
2. Changes to the Fourth 10-Year Inservice Test Interval Relief Request Applicable Code Requirements

C D. Dorman, Regional Administrator - NRC Region I
J. Poole, Project Manager - Hope Creek, USNRC
NRC Senior Resident Inspector - Hope Creek
P. Mulligan, Chief, NJBNE
Tom MacEwen, Hope Creek Commitment Coordinator
Lee Marabella, Corporate Commitment Coordinator

Attachment 1

PSEG Response to Request for Additional Information Regarding Relief Requests VR-01 and VR-02, Associated With the Fourth 10-Year Inservice Test Interval

VR-01 RAI-1:

Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(f), "Inservice testing requirements," requires, in part, that IST of certain American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda, except where alternatives have been authorized pursuant to paragraphs 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

In proposing alternatives, a licensee must demonstrate that the proposed alternatives provide an acceptable level of quality and safety (10 CFR 50.55a(z)(1)) or that compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety (10 CFR 50.55a(z)(2)).

VR-01 requests an alternative for testing excess flow check valves (EFCVs). The alternative is to test EFCVs at a frequency specified in Technical Specifications Surveillance Requirement (SR) 4.6.3.4. SR 4.6.3.4 allows a "representative sample" of EFCVs to be tested every refueling outage such that each EFCV will be individually tested approximately every 10 years. Justification for the relief request is based on General Electric (GE) Topical Report (TR) NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000. The TR provided: (1) an estimate of steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close, and (2) assessment of the radiological consequences of such a release. The NRC staff reviewed the GE TR and issued its safety evaluation on March 14, 2000 (ADAMS Accession No. ML003691722). In its evaluation, the staff found that the test interval could be extended up to a maximum of 10 years. In conjunction with this finding, the NRC staff noted that each licensee that adopts the relaxed test Enclosure 1 interval program for EFCVs must have a failure feedback mechanism and corrective action program (CAP) to ensure EFCV performance continues to be bounded by the TR results. Also, each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the TR.

Please respond to the following:

A) Explain the HCGS failure feedback mechanism and the CAP.

PSEG Response

Excess flow check valve testing procedures contain guidance for the appropriate steps to be taken in response to a test failure. If any EFCV fails its functional test, a notification is entered into the CAP program. Testing scope is expanded to include 2 additional EFCVs from a population scheduled for a future refuel outage. If 1 of 2 of the additional EFCVs fails its functional test, then a second notification is entered into the CAP program to document the failure. In addition, the Maintenance Rule program is used to track the performance of the excess flow check valves. At Hope Creek, EFCVs are condition monitored to comply with paragraph (a)(2) of the Maintenance Rule. If failures are discovered, they are evaluated as “preventable system functional failures” in the corrective action program. Maintenance Rule a(1) status is entered based on repeat system functional failures of an EFCV, and the performance or condition of the EFCV shall be monitored in a manner sufficient to provide reasonable assurance that it is capable of fulfilling its intended function.

B) Explain how the CAP evaluates component failures and what appropriate corrective actions would likely be taken.

PSEG Response

Operations shift management reviews all CAP notifications which involve a condition adverse to quality, including component failures, to ensure the appropriate corrective actions are taken. After Operations shift management screens the notification and takes the appropriate actions required by the Technical Specifications, the notification is screened by the Station Ownership Committee. The significance level and evaluation type for the notification are assigned based upon the impact of the condition, as defined by the procedure for issue identification and screening. Acceptance criteria for valve operability are provided in the applicable test procedures and the completed test packages are reviewed by the IST program engineer.

The initial corrective action for an EFCV failure would be expanded testing scope and evaluation in accordance with the requirements of 10 CFR 50.65 as described above.

C) Explain the radiological dose assessment and release frequency analysis, confirming that they bound the generic analyses of GE TR NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.

PSEG Response

The radiological consequences for an instrument line break have been evaluated in Updated Final Safety Analysis Report (UFSAR) Section 15.6.2.5. The analysis does not credit the EFCVs for isolating the break and assumes a discharge of reactor water through an instrument line with a ¼ inch restricting orifice throughout the event. The analysis confirms that the radiological consequence of EFCVs failing to function upon demand is sufficiently low to be considered insignificant.

The calculations contained in GE TR NEDO-32977-A 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 1.01 E-7/hr. At Hope Creek, there were no EFCV failures in over 5 years of testing experience for 105 valves (525 valve operating years), resulting in a failure rate of 0 failures/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.

Hope Creek has failure rates consistent with the results of the GE TR NEDO-32977-A. Seven plants reported no failures of EFCVs during the operating period, while the remaining 5 plants reported between 1 and 4 failures.

VR-02 RAI-1:

Title 10 of the Code of Federal Regulations Section 50.55a(f), "Inservice testing requirements," requires, in part, that IST of certain American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda, except where alternatives have been authorized, pursuant to paragraphs 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

In proposing alternatives, a licensee must demonstrate that the proposed alternatives provide an acceptable level of quality and safety (10 CFR 50.55a(z)(1)) or that compliance would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety (10 CFR 50.55a(z)(2)).

Please provide information to demonstrate that testing the Target Rock main bodies at a five year interval as required by OM Code, but not staggering the testing of the main bodies through the five year interval (which is also required by OM Code) will provide an acceptable level of quality and safety.

A) In Section 5 of the relief request, "Proposed Alternative and Basis for Use," the following statement is made:

> Testing of the main body (mechanical portion), which contains only the main disc, piston rings and a preload spring that is non-adjustable, at the Mandatory Appendix I specified frequency will not result in a significant increase in the level of safety.

In Section 4 of the relief request, "Reason for Request," the following statement is made:

> The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

Provide further justification for these statements, especially in regard to the three stage Target Rock Model 0867F valves, which have sustained main body damage and degradation over the course of just a single fuel cycle at some plants.

PSEG Response

Testing of the main body (mechanical portion) of three stage Target Rock Model 0867F valves, if installed, will be performed at the Mandatory Appendix I specified frequency.

The Target Rock 2-stage main valves have demonstrated good performance. In the past five years (3 refuel cycles), only one Safety Relief Valve (SRV) main body has experienced an issue which resulted in significant main seat leakage. In September 2014, PSEG replaced SRV-H main body and pilot valve. A leakage cause analysis was performed. No significant binding or looseness was noted during disassembly, and no evidence of any other damage was observed beyond the steam cutting of the main disc and seat. The main spring was tested satisfactorily.

B) Provide further information on how and when the various discrete tests listed in OM Code, Mandatory Appendix I, paragraph I-3310, will be accomplished for the subject valves.

PSEG Response

The various discrete tests listed in OM Code, Mandatory Appendix I, paragraph I-3310, will be accomplished as described below for Target Rock two-stage Safety/Relief Valves.

OM Code, Mandatory Appendix I, paragraph I-3310 main valve tests and inspections are performed once every five years, typically at a PSEG-approved vendor facility:

(a) Visual examination

(b) Seat tightness determination, if practicable

(i) Determination of compliance with the Owner's seat tightness criteria

OM Code, Mandatory Appendix I, paragraph I-3310 pilot valve tests and inspections are performed once every 18 months, typically at a PSEG-approved vendor facility:

(a) Visual examination

(b) Seat tightness determination, if practicable

(c) Set pressure determination

(d) Determination of electrical characteristics and pressure integrity of solenoid valve(s)

(e) Determination of pressure integrity and stroke capability of air actuator

(h) Determination of actuating pressure of auxiliary actuating device sensing element, where applicable, and electrical continuity

(i) Determination of compliance with the Owner's seat tightness criteria

The following tests are not applicable to Hope Creek Target Rock 2-stage Model 7567F SRVs:

(f) Determination of operation and electrical characteristics of position indicators

(g) Determination of operation and electrical characteristics of bellows alarm switch

Attachment 2

Changes to the Fourth 10-Year Inservice Test Interval Relief Request Applicable Code Requirements

The proposed alternatives in Reference 1 were to the requirements of the ASME OM - 2012 Edition, based on PSEG's understanding of the timetable to complete the current rulemaking to incorporate the 2012 Edition by reference into 10 CFR 50.55a. However, due to uncertainty as to when the final rule will be issued, PSEG is electing to change the applicable code edition to the 2004 Edition with 2006 Addenda, which is the latest edition and addenda of the OM Code currently incorporated by reference in 10 CFR 50.55a(a)(1)(iv). Resulting changes in the applicable code requirements for each proposed alternative are described below.

1. 10 CFR 50.55a Request GR-01

   The following code requirements are deleted:

   - Appendix II, II-4000(b)(1)(g)
   - Appendix III, III-3310(b)
   - Appendix III, III-3310(c)
   - Appendix III, III-3722(c)
   - Appendix III, III-3722(d)

   The following requirements are added:

   - Appendix II, II-4000(b)(1)(e) -Condition-Monitoring Activities, Optimization of Condition-Monitoring Activities; "Intervals shall not exceed the maximum intervals shown in Table II-4000-1." Table II-4000-1 lists three intervals -10, 12, and 16 years.
   - OMN-1 (2006 Addenda), 3.3.1(b) - Inservice Test Interval; "...MOV inservice testing shall be conducted every 2 refueling cycles or 3 years (whichever is longer)..."
   - OMN-1, 3.3.1(c) - Inservice Test Interval; "The maximum inservice test interval shall not exceed 10 years."
   - OMN-1, 3.6.1 - Normal Exercising Requirements; "...with the maximum time between exercises to be not greater than 24 months."
   - OMN-1, 3.7.2.1 - HSSC MOVs; "HSSC MOVs that can be operated during plant operation shall be exercised quarterly, unless..."
   - OMN-1, 3.7.2.2(c) - LSSC MOVs; "...using an initial test interval of three refueling cycles or 5 years (whichever is longer)..."
   - OMN-1, 3.7.2.2(d) - LSSC MOVs; "LSSC MOVs shall be inservice tested at least every 10 years..."

2. 10 CFR 50.55a Request PR-01

   No changes are required to the applicable code requirements.

3. 10 CFR 50.55a Request PR-02

   No changes are required to the applicable code requirements.

4. 10 CFR 50.55a Request VR-01

   No changes are required to the applicable code requirements.

5. 10 CFR 50.55a Request VR-02

   No changes are required to the applicable code requirements.