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10 CFR 50.55a

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June 11, 2008

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

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

Subject: Response to Request for Additional Information Related To Relief Requests HC-I3R-01 and HC-I3R-02 (TAC Nos. MD7503 and MD7504)

Reference:
1) Letter from Jeffrie Keenan (PSEG Nuclear LLC) to USNRC, December 12, 2007
2) Letter from USNRC to William Levis (PSEG Nuclear LLC), May 7, 2008

In Reference 1, PSEG Nuclear LLC (PSEG) requested relief from certain requirements specified in Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for the third inservice inspection interval at Hope Creek Generating Station. In Reference 2, the NRC requested additional information. Attachment 1 to this letter provides the requested information. Attachment 2 summarizes regulatory commitments associated with PSEG's response.

Should you have any questions regarding this submittal, please contact Mr. Paul Duke at 856-339-1466.

Sincerely,

Christine T. Neely
Director - Regulatory Affairs

A047
NRR

Attachments:
1. Response to Request for Additional Information
2. Summary of Regulatory Commitments

cc: S. Collins, Regional Administrator – NRC Region I
R. Ennis, Project Manager - Hope Creek, USNRC
NRC Senior Resident Inspector - Hope Creek
P. Mulligan, Manager IV, NJBNE

ATTACHMENT 1

Hope Creek Generating Station

Facility Operating License No. NPF-57
NRC Docket No. 50-354

Response to Request for Additional Information Related to Relief Requests HC-I3R-01 and HC-I3R-02 (TAC Nos. MD7503 and MD7504)

In Reference 1, PSEG Nuclear LLC (PSEG) requested relief from certain requirements specified in Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for the third inservice inspection interval at Hope Creek Generating Station. In Reference 2, the NRC requested additional information.

PSEG's responses are provided below.

Relief Request HC-I3R-01

1. Per Regulatory Guide (RG) 1.193, Revision 2 (October 2007), Code Case N-578-1 is listed as an unacceptable Section XI Code Case. Please provide a justification for the application of Subarticle -2430 of Code Case N-578-1. Explain how the use of Code Case N-578-1 provides a "more refined methodology for implementing necessary additional examinations."

   Response
   The "more refined methodology for implementing necessary additional examinations" refers to Code Case N-578-1, Subarticle -2430, "Additional Examinations," when compared to the high level discussion in Electric Power Research Institute (EPRI) TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A.

   EPRI TR-112657, section 3.6.6.2, has a brief discussion of additional examinations under the context of an evaluation, with little detail regarding the evaluation process. Subarticle -2430 of the code case uses a similar method but provides a more descriptive process based on postulated failure mode and impact of failure potential. The code case also adds a second expansion process should further flaws or relevant conditions be found in the first expanded scope, as well as providing guidance for returning the components receiving additional examinations back into the normal periodic schedule.

2. Have any welds that were selected for inspection in the Risk-Informed Inservice Inspection (RI-ISI) program that was approved by the NRC staff in Reference 1 been removed from the population of welds that will be inspected during the third ten-year interval? If so, why were the welds removed from the population of welds to be inspected?

Response
Yes, there are changes to weld selection based on accessibility, personnel radiation exposure, and configuration changes. Changes to the Risk-Informed Inservice Inspection (RISI) Program have been summarized in one comprehensive table and reply. See the response to Question 3 below for this information.

3. Have any welds that were not selected for inspection in the RI-ISI program that was approved by the NRC staff in Reference 1 been selected for inspection during the third ten-year interval? If so, why were the welds added to the population of welds to be inspected?

Response
Yes, there are changes to weld selection based on accessibility, personnel radiation exposure, and configuration changes.

For the third interval, the overall scope of the program is similar to the second interval. No new systems or modifications to how the methodology was applied have been made that affect the program scoping process. However, the RISI program is required to and has been maintained as a living program assessing component and configuration changes and major PRA model revisions. The following table summarizes the welds to be examined in the RISI Program and identifies the changes to the RISI populations for HCGS as part of the living program process:

RISK RANK	EXAMS (RISI REV. 1)	EXAMS (RISI REV. 2)	ITEMS AFFECTING CHANGES
High	19	23	▪ Limited Exam Coverage ▪ Plant/Component Modifications ▪ PRA Model Revisions[1] ▪ Extended Power Uprate (EPU)
Medium	85	84	▪ Limited Exam Coverage ▪ Plant/Component Modifications ▪ PRA Model Revisions[1] ▪ Extended Power Uprate (EPU)
Total	**104**	**107**	
[1] Latest incorporated revision is PRA Model 2.0 with EPU considerations.			

Limited Exam Coverage – The examinations selected were modified in some cases to optimize examination code coverage.

Plant Modifications – Some of the larger scope plant modifications that were made include Extended Power Uprate, Reactor Pressure Vessel Head Spray line deletion, and removal of post accident sampling system (PASS).

PRA Model Revisions – The PRA Model was revised twice before the changes were incorporated in this update of the RISI Program. The first update in 2003 incorporated recent operating experience, changes to key plant procedures, and changes in operator training, system success criteria and industry and plant-specific event and failure data. The second update in 2004 addressed the effect of EPU, including physical modifications, setpoint changes and procedure changes.

Extended Power Uprate – Extended Power Uprate license change was issued May 14, 2008 (TAC No. MD3002) and increases authorized maximum power level by approximately 15 percent from the previous licensed thermal power of 3,339 megawatts thermal to 3,840 megawatts thermal.

4. The relief request states that:

> The Risk Impact Assessment completed as part of the original baseline RISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the Third Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RISI methodology and evaluation will be maintained for the new interval. As such, the original risk impact assessment process is not impacted by the new interval and does not require update.

The NRC staff does not concur with the implication that, if there is no change in methodology, the change in risk assessment is not part of the living process. RG 1.178, Standard Review Plan (SRP) 3.9.8, and Electric Power Research Institute (EPRI) Topical Report TR-112657 (References 2, 3, and 4) require an evaluation of the change in risk arising from the proposed change in the ISI program. Please provide a discussion on the potential change in risk between the RISI program proposed for implementation in the third interval and the ASME Section XI requirements from which relief was granted in Reference 1. If inspections were discontinued or relocated between the second and third intervals' RISI programs, please provide an estimate of the change in risk.

Response

The original risk impact assessment was issued in Revision 1 of the HCGS RISI Evaluation. For the Third ISI Interval, the methodology of the calculation has not changed, and the calculation remains a part of the living program. In maintaining this portion of the RISI Evaluation living, the change-in-risk continues to be assessed against the pre-risk-informed 1989 ASME Section XI Program.

Using this process, the change in core damage frequency (CDF) is currently 7.41E-09/year and the change in large early release frequency (LERF) is 7.44E-10/year. In Revision 1 to the RISI Evaluation the change in CDF was

1.21E-09/year and the change in LERF was 1.21E-10/year. These values are within the 1.00E-06 and 1.00E-07 acceptance criteria for delta CDF and delta LERF, respectively. The change-in-risk analysis was likewise done at a system level, and the system acceptance criteria were not exceeded for any individual system within the RISI Program.

5. The relief request states that:

> As an added measure of assurance, any new systems, portions of systems, or components being included in the RISI Program for the Third Interval will be added to the Risk Impact Assessment performed during the previous interval. These components will be addressed within the evaluation at the start of the new interval to assure that the new Third Interval RISI element selection provides an acceptable overall change-in-risk.

The results of the evaluations described above should be part of a request for relief to support the required finding that the proposed program provides an acceptable level of quality and safety. Please provide a brief description of these evaluations and an overview of the results.

<u>Response</u>
The RISI evaluation current for the Third Interval is Revision 2 as discussed further in the responses to Questions 3 and 6. The net changes to the RISI element selection are summarized in Response 3 and the current change-in-risk impact as a result is provided in Response 4. Since that revision, no major modifications have been made affecting the RISI program scope, and since the original program approval, no new systems have been added or removed from the boundaries subject to the RISI evaluation process.

6. The relief request states that:

> These portions of the RISI Program have been and will continue to be reevaluated and revised as major revisions of the site PRA [probabilistic risk assessment] occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, and Element Selection steps encompass the complete living program process...

Please provide the date of the last reevaluation and revision that is described above and a brief description of the results of the reevaluations and revisions undertaken at that date.

Response
The most recent reevaluation updated the RISI Evaluation to Revision 2 as discussed above. This revision was completed early in 2006 to incorporate the changes to PRA and changes resulting from implementing an extended power uprate. The revision included updates to the Consequence and Degradation Mechanism Evaluations, Risk Ranking, Element Selection and the Risk Impact Analysis.

As a result of the degradation mechanism update, feedwater welds moved from risk category 4 to risk category 2 due to thermal transient (TT), and two Reactor Recirculation system welds were assigned intergranular stress corrosion cracking (IGSCC). As a result of the consequence evaluation update, some Low Pressure Coolant Injection (LPCI), Core Spray, Standby Liquid Control (SLC), Reactor Water Cleanup (RWCU), and Reactor Core Isolation Cooling (RCIC) system welds increased from low consequence to medium consequence (Risk Category 7 to Risk Category 6). The revised high and medium Risk Rankings drive the examination populations summarized in Response 3.

References:

1) Letter from Darrell J. Roberts (NRC) to A. Christopher Bakken, III (PSEG) dated December 8, 2004, "Hope Creek Generating Station - Implementation of a Risk-Informed Inservice Inspection Program" (ADAMS Accession No. ML043080161)

2) Regulatory Guide 1.178, September 2003, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping" (ADAMS Accession No. ML032510128)

3) NUREG-0800, SRP Chapter 3.9.8, September 2003, "Standard Review Plan for the Review of Risk-Informed Inservice Inspection of Piping" (ADAMS Accession No. ML032510135)

4) EPRI Topical Report TR-112657, Revision B-A, January 2000, "Revised Risk-Informed Inservice Inspection Evaluation Procedure" (ADAMS Accession No. ML013470102)

Relief Request HC-I3R-02

1. Article IWF-5000, Subsections IWF-5200(c) and IWF-5300(c) clearly state that integral and non-integral attachments for snubbers (including lugs, bolting, pins, and clamps), shall be examined in accordance with the requirements of Subsection IWF. Please explain whether and how these requirements will be met.

Response
Visual inspections in accordance with Surveillance Requirement (SR) 4.7.5.c shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Insulation or interferences are not required to be removed unless discoloration or distortion are evident adjacent to snubber attachment hardware.

The applicable visual inspection guidelines do not differentiate between integral and non-integral attachments. Visual examinations include verification that attachments to foundation or supporting structure are secure and a check for any evidence of pipe clamp movement (walking or rotation). This provides an acceptable level of quality and safety in lieu of the requirements of Article IWF-5000, Subsections IWF-5200(c) and IWF-5300(c).

2. The relief request and HCGS Technical Specification (TS) 3/4.7.5 do not address the requirements of ASME/American Nuclear Standards Institute (ANSI) *Code for Operation and Maintenance of Nuclear Power Plants* (OM), Part 4 (OM-4), Section 3.2.4, specifically Section 3.2.4.2, "Test Failure Mode Groups." Please explain how TS 3/4.7.5 meets these requirements.

Response
OM-4 paragraph 3.2.4.2 requires unacceptable snubber(s) to be categorized into test failure mode group(s). A test failure mode group(s) shall include all unacceptable snubbers that have a given failure mode, and all other snubbers subject to the same failure mode. The following failure modes shall be used:

(a) design/manufacturing
(b) application induced
(c) maintenance/repair/installation
(d) isolated
(e) unexplained

OM-4 paragraph 3.2.5.2(c) permits failure mode groups to be separated for continued testing from the general population of snubbers. However, at least one additional random sample from the general population is required to be tested. Any additional failures within the failure mode group are counted for continued testing only for that failure mode group.

SR 4.7.5.g requires an engineering evaluation of each functional test failure to determine the cause of the failure. The results of this evaluation are used in selecting additional snubbers for testing in accordance with SR 4.7.5.f. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.5.e for snubbers not meeting the functional test acceptance criteria. Thus, for failures due to design/manufacturing, the TS 3/4.7.5 requirements are essentially the same as those of OM-4, paragraph 3.2.4.2.

For other failure modes, because TS 3/4.7.5 does not have specific allowances for failure mode grouping, a more conservative additional sample from the overall population can result. For all failure modes, SR 4.7.5.g ensures that snubbers subject to the same failure mode are selected for continued testing.

3. Surveillance Requirement (SR) 4.7.5.e.2, provides an optional functional testing of snubbers in accordance with TS Figure 4.7.5-1. Please verify that: (1) this plan is equivalent to the 37 testing sample plan of OM-4; and (2) explain whether and how the requirement of additional sampling of at least one-half the size of the initial sample lot as required by OM-4, Paragraph 3.2.3.2(b), will be met while using TS Figure 4.7.5-1.

<u>Response</u>
The sample plan in SR 4.7.5.e.2 is equivalent to the 37 testing sample plan of OM-4. An initial random sample of 37 snubbers of each type is chosen and functionally tested. As defined in SR 4.7.5.a, snubber type means snubbers of the same design and manufacture, irrespective of capacity.

"C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of testing "N" snubbers, the results shall be plotted on Figure 4.7.5-1. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. The "Accept" line in Figure 4.7.5-1 uses the equation C=0.055N-2.007.

If one snubber fails to meet the acceptance criteria, then an additional sample of 19 snubbers of the failed type selected in accordance with SR 4.7.5.g will be tested. If a second snubber fails to meet the acceptance criteria then an additional ample of 18 snubbers of the failed type will be tested. This testing continues until the number of failed snubbers falls below the "Accept" line or all the snubbers of that type have been tested. Thus, the requirement of additional sampling of at least one-half the size of the initial sample lot as required by OM-4, Paragraph 3.2.3.2(b), will be met while using TS Figure 4.7.5-1.

4. SR 4.7.5.c, "Visual Inspection Acceptance Criteria," states, in part, that "the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.4.f." Please confirm that the reference to "Specifications 4.7.4.f" is in error and should actually be "Specification 4.7.5.f." If SR 4.7.5.c is in error, please enter this item into the corrective action program and provide a regulatory commitment to request a license amendment to correct the error.

   Response
   The reference to Specifications 4.7.4.f is a typographical error and has been entered into PSEG's corrective action program. A regulatory commitment to submit a license amendment request correcting the error is provided in Attachment 2 to this letter.

References

1) Letter from Jeffrie Keenan (PSEG Nuclear LLC) to USNRC, December 12, 2007

2) Letter from USNRC to William Levis (PSEG Nuclear LLC), May 7, 2008

ATTACHMENT 2

Hope Creek Generating Station

Facility Operating License No. NPF-57
NRC Docket No. 50-354

Response to Request for Additional Information Related to Relief Requests HC-I3R-01 and HC-I3R-02 (TAC Nos. MD7503 and MD7504)

Summary of Commitments

The following table identifies commitments made in this document. (Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		One-Time Action (Yes/No)	Programmatic (Yes/No)
PSEG will submit a license amendment request correcting the typographical error in Surveillance Requirement 4.7.5.c	12/31/2008	Yes	No