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LR-N02-0413
LCR H02-013

PSEG
Nuclear LLC

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

Gentlemen:

HOPE CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION REGARDING INTEGRATED LEAK RATE TEST INTERVAL EXTENSION (TAC NO. MB6551)
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Reference: Letter LR-N02-0319, *Request for One-Time Extension to Increase the Interval of the Integrated Leak Rate Test from Ten to Twenty Years,* dated October 9, 2002

On October 9, 2002 PSEG Nuclear LLC (PSEG) submitted the referenced request for a revision to the Technical Specifications (TS) to extend the Type A Containment Integrated Leak Rate Test (ILRT) in Section 6.8.4.f from once per 10 years to once per 20 years for the Hope Creek Generating Station. On November 22, 2002 PSEG resubmitted LCR H02-013 to request an extension to 15 years, 5 years less than the original request.

In a letter dated November 22, 2002, PSEG received a request from the NRC for additional information regarding integrated leak rate test interval extension at Hope Creek Generating Station. This request for additional information was discussed with Mr. George Wunder, NRC Hope Creek Project Manager and other members of the NRC on November 21, 2002. Attachment 1 contains PSEG's responses.

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

Sincerely,

Gabor Salamon
Manager – Nuclear Safety and Licensing

Attachment

A017

C: Mr. H. Miller, Administrator – Region I
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HOPE CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION REGARDING INTEGRATED LEAK RATE TEST INTERVAL EXTENSION (TAC NO. MB6551)

NRC Question 1:

Your request for amendment dated October 9, 2002, does not describe the containment Inservice Inspection (ISI) program being implemented at Hope Creek Unit 1. Please provide a description of the ISI methods (with Code Edition and addenda) that provide assurance that in the absence of an Integrated Leak Rate Test (ILRT) for 15 years, the containment structural and leak tight integrity will be maintained. Also provide start and completion dates of the first IWE and IWL examinations performed as required by 10CFR50.55a, and a schedule for conducting the future examinations.

Response:

A general visual inspection is performed on the Hope Creek Generating Station (HCGS) containment in accordance with Section XI sub-section IWE of the 1998 Edition of the ASME Code including the 1998 Addenda. Use of this edition of the Code was requested through PSEG Nuclear LLC (PSEG) relief request RR-E1 (LR-N99-0409) dated October 7, 1999 and authorized by NRC letter dated June 6, 2000. The inspection interval was established such that it coincided with the Hope Creek inspection interval for sub-sections IWA, IWB, IWC, IWD and IWF of ASME Section XI. The interval start date was December 13, 1997 and is scheduled to complete after RFO13 scheduled for Spring, 2006. After refueling outage 13 (RFO13), the program will be upgraded in accordance with 10CFR50.55a for the next 120-month inspection interval (Third Inspection Interval).

All required IWE examinations were completed during refueling outage 9 (RFO9) that also concluded the first inspection period for the interval. The areas and items subject to examination included the accessible containment surface areas, including structural attachments and penetrations, pressure retaining bolting and class MC supports.

NRC Question 2:

IWE-1240 requires licensees to identify the surface areas requiring augmented examinations, Please provide the NRC staff with the list of the areas (such as shell near sand cushion areas and vertical portions of the drywell) identified for augmented examination and a summary of examinations performed.

Response:

The HCGS Containment Inservice Inspection (CISI) Program does not currently contain any surface areas classified as Augmented Exam areas in accordance with Table IWE-2500-1, Examination Category E-C. This determination was based on an initial technical position developed to document the results of evaluations conducted against the requirements of paragraph IWE-1240 of the ASME B&PV Code, Section XI, 1998 Edition, including 1998 Addenda.

The technical position was based upon a review of industry experience documents, HCGS plant specific experience, two formal PSEG Engineering Evaluations, identification of the design corrosion allowances, and review of a specific evaluation documenting the condition of the applied surface coatings within the primary containment. The Technical Position document and supporting Engineering Evaluations are on file at the plant location.

As a result of the visual examinations performed to date, no surface areas have been identified requiring re-classification in accordance with Paragraph IWE-1240 as augmented examination areas subject to the requirements of Table IWE-2500-1, Examination Category E-C.

Table 1 gives an approximation of both the accessible and inaccessible areas of the primary containment. The inaccessible areas are not inspected as part of the visual exam.

DRYWELL		
	Accessible	**Inaccessible**
Inside	~97%	~3%
Outside	~5%	~95%
TORUS		
	Accessible	**Inaccessible**
Inside	~98%	~2%
Outside	~99%	~1%

NRC Question 3:

For the examination of seals and gaskets, and examination and testing of bolts associated with the primary containment pressure boundary (examination categories E-D and E-0, IWE-1992), you have previously requested relief from the IWE-1992; this relief was authorized by NRC letter dated June 6, 2000. As an alternative to IWE-1992 you planned to examine these components during leak rate testing of the primary containment. With the flexibility provided in Option B of Appendix J for Type B and Type

C testing, and the extension requested in this amendment for Type A testing, please provide the schedule for examination and testing of seals, gaskets, and bolts that provide assurance of the integrity of the containment pressure boundary.

Response:

The 1998 edition of ASME Section XI subsection IWE does not require any examination of seals and gaskets. Under the 10CFR50 Appendix J, Option B program those Type B penetrations that utilize resilient seals, gaskets, etc. are tested within the guidelines provided by Option B and Regulatory Guide 1.163. Most of the Type B penetrations at HCGS are on an extended test frequency based on performance with a percentage of the total population tested each refueling outage. This program is set-up such that 100% of all components are tested during a running 10-year interval. Those components that do not fall under extended test frequencies are tested at least once every 30 months. These components are either on penetrations that are disassembled and reassembled each outage or have not demonstrated acceptable performance history per the Primary Containment Leakage Rate Testing Program. Components that are disassembled and reassembled during an outage receive an as-found test prior to any work and an as-left test after all work is completed. The gasket or other sealing material is inspected prior to closing and as-left testing.

All pressure retaining bolting is examined at least once each inspection period, as scheduled in the ISI Long Term Plan, either in place or removed. A connection will not be disassembled solely in support of this examination. If the bolting is found to be outside the general visual acceptance criteria, then a detailed examination (VT-1, or equivalent) will be performed on the bolting. If the potentially degraded bolting is assembled, then it will be disassembled to facilitate the detailed examination.

If a bolted connection is disassembled at the time of the inspection all accessible areas shall be visually examined (VT-3 or VT-1, if necessary). Pressure retaining containment bolted connections that are disassembled and not scheduled to be examined by ASME Section XI, are examined using PSEG procedures based on normal maintenance procedures using their professional training and the skill of the craft. When an IWE boundary component is disassembled then reassembled for the maintenance activities, an Appendix J local leak rate test would be performed to determine the leak-tight integrity of the component.

NRC Question 4:

The stainless steel bellows have been found to be susceptible to trans-granular stress corrosion cracking, and leakages through them are not readily delectable by Type B testing (see NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing). In general, boiling water reactor Mark I primary containments have bellows on the vent lines between the drywell and the torus, as well as on several process piping penetrations in the drywell. If degraded, the bellows could allow the drywell steam and air to bypass the suppression pool during loss-of-coolant accidents and core damage accidents. Please provide information regarding inspection and testing of the bellows at Hope Creek.

Response:

Bellows assemblies are examined per the requirements of the IWE program. They are also local leak rate tested as part of the 10CFR50 Appendix J Option B program. Although most of the bellows assemblies are eligible for extended frequency testing, PSEG has conservatively elected to test them every other refueling outage with approximately 1/2 of the assemblies tested each outage. This frequency was established for two reasons; the first was due to the potential concerns in Information Notice 92-20 and the second was to ensure that the assemblies met the ASME Section XI subsection IWC system pressure test requirement of being examined at least once each inspection period.

There are two bellows assemblies at HCGS that have been identified with minor leakage. The first one was discovered in March 1994. Extensive examination and troubleshooting was performed on this assembly to determine the source of the leakage but no source of external leakage was noted. It is suspected that the leakage is minor thru wall leak(s) on the inner ply of this assembly. This same bellows assembly was also tested as part of the Integrated Leak Rate Test performed at the end of this outage and no new source of external leakage was located. This assembly, because of a previously identified leak is not eligible for extended frequency testing and will continue to be tested each refueling outage. The leakage has remained unchanged since 1994. A second bellows assembly was found with a minor leak in December 1995. No source of external leakage could be located on this assembly. This bellows is also tested each refueling outage and the results have also remained unchanged since the 1995 test. There is a design change package that was developed to encapsulate the affected bellows if a significant change in the leakage were to occur. This same package would be utilized if a general and detailed visual examination revealed degradation that made the leak tight integrity of the assembly suspect.

NRC Question 5:

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, and Oyster Creek), have indicated degradation from the uninspectable side of the steel shell and liner of primary containments. The major uninspectable areas of the Mark I containment are the vertical portion of the drywell shell and part of the shell sandwiched between the drywell floor and the basemat. Please discuss what programs are used to monitor their condition. Also, address how potential leakage due to age related degradation from these uninspectable areas is factored into the risk assessment in support of the requested ILRT interval extension.

Response:

Under the IWE program the acceptability of inaccessible areas will be evaluated if conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to the inaccessible area. This evaluation shall include the description of type, estimated extent, cause of the degradation, evaluation and results of each area and description of necessary corrective actions.

Our submittal of October 9, 2002 did not address potential leakage due to age related degradation in uninspectable areas. We also did not take any credit for the visual inspections (IWE) of the inspectable areas. In our November 21, 2002 telecon, the NRC referred to a risk assessment performed by Calvert Cliffs (CCNPP) which addresses the potential for leakage due to age related degradation for uninspectible areas. Our response is based upon the method developed by CCNPP.

The method developed by CCNPP made several assumptions to support a six-step calculation. Step 6 derives the likelihood of non-detected containment leakage. Assumptions important to the CCNPP method:

1. The success data was tied to the initiation of IWE. (The beginning point of the statistics.)
2. The liner flaw likelihood is doubled every five years. (The basic assumption of aging.)
3. The IWE effectiveness is 90% in the inspectable region.

Another assumption that was not important to the CCNPP method was the partition of regions (Containment Cylinder and Dome, and Containment Basemat). The partition provides two sets of boundary conditions. Therefore, two sets of degradation results are obtained. However, the partition itself (%) was never entered into the final calculation. In other words, regardless of the size of the uninspectable region, the numerical results do not change.

The following is a step-by-step discussion following the CCNPP method:

1. In terms of the actual calculation, steps 1 through 3 of the CCNPP method are generic. The increase in Flaw Likelihood Between 3 and 15 Years for HCGS is assumed to be the same as CCNPP, namely, 8.7% for the Containment Cylinder and Dome, and 2.2% for the Containment Basemat.

2. Step 4 uses a plant specific assumption to obtain one anchor point for a lognormal distribution. For HCGS, the 100% likelihood of containment breach corresponds to a pressure of 170 psia. CCNPP's breach pressure is assumed to be 150 psia. To be conservative PSEG will use the CCNPP number, 1.1%.

3. Step 5 is a generic assumption and is adopted by HCGS. The likelihood of failure to detect a leak in an inspectable area is 10%.

4. Step 6 derives the Likelihood of Non-Detected Containment Leakage by multiplying the results from steps 3, 4, and 5. Since HCGS took a conservative approach, the derived Likelihood of Non-Detected Containment Leakage is the same as CCNPP, namely 0.0096% for the Containment Cylinder and Dome and 0.0024% for the Containment Basemat.

In order to derive the impact on large early release frequency (LERF), a more accurate approach would be to weight the values derived in Step 6 by their relative percentage of area then sum them. CCNPP took a very conservative approach mathematically. By not weighting the area, it essentially assumed two containments, one inspectable and the other not. Using the CCNPP methodology, the Total Likelihood of Non-Detected Containment Leakage for HCGS is 0.012%. Therefore, the increase in LERF (ILRT 3 to 15 years) = 0.012%*8.89E-6 = 1.1E-9 per year.

Based upon the above comparison there will be no meaningful impact to the original HCGS analysis. In addition, this is a very conservative number as a number of conservative assumptions were used for derivation. Furthermore, the HCGS calculation did not take credit for the IWE. To apply the CCNPP methodology for accounting for the uninspectable area for HCGS, it would be necessary to take credit for the IWE, and then apply the above estimated correction. This would result in an even smaller value than the one submitted in our letter of October 9, 2002. Therefore, no further calculations were performed.