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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-001

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

Subject: Report of Changes, Tests, and Experiments

Pursuant to the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments," paragraph (d)(2), Hope Creek Generating Station (HCGS) is providing the required report (Attachment 1) for Renewed Facility Operating License No. NPF-57. This report provides a summary of 10CFR50.59 evaluations for activities implemented under 10CFR50.59 at HCGS during the period of January 1, 2017, through December 31, 2018.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Thomas cachaza at 856-339-5038.

Sincerely,

David Mannai
Senior Director - Regulatory Operations
PSEG Nuclear LLC

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Attachment 1: Report of Changes, Tests, and Experiments

cc: Mr. David Lew, Regional Administrator - NRC Region 1
Mr. James Kim, Project Manager - USNRC
Mr. Justin Hawkins, USNRC Senior Resident Inspector - Hope Creek
Mr. Patrick Mulligan, Manager IV, NJBNE
Mr. Lee Marabella, Corporate Commitment Tracking Coordinator
Mr. Thomas Cachaza, Hope Creek Commitment Tracking Coordinator

Attachment 1
Hope Creek Generating Station
Renewed Facility Operating License NPF-57
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Report of Changes, Tests, and Experiments

Replacement of the Reactor Recirc Pump MG Sets with Siemens VFD

The Hope Creek Reactor Recirculation Pump (RRP) Motor-Generator (MG) Sets were replaced with Siemens Variable Frequency Drives (VFD) in April 2018 during refuel outage 21. The Hope Creek RRP MG Sets provided variable frequency drive power to the RRP motors to change Recirc flow. The Siemens VFDs are a series of cells linked together to create the medium voltage power output of the drive system. The VFD system controls the speed of the Reactor Recirculation Pump (RRP) by adjusting the frequency of the electrical power supplied to the motor. A 50.59 evaluation was performed due to the determination that the proposed activity involved a change to an SSC that adversely affected a UFSAR described design function. This conclusion was based upon the change from analog to digital controls.

The 50.59 evaluation utilized NRC Information Notice 2010-10 for guidance regarding non-safety related digital upgrades. Because control of the Reactor Recirculation Pumps can directly impact core reactivity and cause a reactor transient including a reactor trip, the RRP digital upgrade modification was evaluated as important to safety. The evaluation concluded that there would be no increase in the probability or consequences of a malfunction, and each of the eight evaluation questions was answered no.

Replacement of one Main Steam Safety Relief Valve (SRV)

One two-stage main steam Safety Relief Valve (SRV) was replaced with a three-stage SRV in April 2018 during refuel outage 21. The replacement three-stage SRV has the same setpoint, capacity and response time as the original two-stage SRV. Hope Creek has experienced setpoint drift of the two-stage SRVs due to corrosion bonding of the pilot disc and seat. The change to three-stage SRVs is being performed to address the issue of setpoint drift. There was no change to Technical Specifications or the Facility Operating License, however the Hope Creek UFSAR discusses the use of two-stage SRVs versus three-stage SRVs due to concerns of spurious opening of the three-stage SRV that existed at the time of original plant design. Due to the early concerns associated with three-stage SRV, the replacement was considered to be a change to an SSC that potentially adversely affected an UFSAR design function.

The NRC issued Generic Safety Issue (GSI) B-55, on the basis of concerns about the three-stage SRVs. The NRC has since issued RIS-2000-12, Resolution of Generic Safety Issue B-55, "Improved Reliability of Target Rock safety Relief Valves." Industry operating experience has demonstrated that new design features of the three-stage SRVs have been effective in preventing inadvertent openings. Based on the combination of industry experience and newer instrumentation installed on the 3-stage models it is concluded that there is no increase in the likelihood of a spurious opening of the 3-stage model versus the 2-stage model. The evaluation concluded that there would be no increase in the probability or consequences of a malfunction, and each of the eight evaluation questions was answered no.

Revision of Dose Calculations to Include Condensate Storage Tank Bypass Leakage

Hope Creek identified additional post-LOCA leakage pathways through Motor Operated Valves (MOVs) located in HPCI and RCIC test lines to Condensate Storage Tank (CST) which could impact the offsite and onsite dose consequences determined in the following dose calculation analyses:

H-1-ZZ-MDC-1880, Rev 7, "Post-LOCA EAB, LPZ, and CR Doses (Ref. 1)"
H-1-ZZ-MDC-1923, Rev 5-A, "Access to Areas Requiring Continuous Occupancy (Ref. 2)"
H-1-ZZ-MDC-1927, Rev 4-A, "Vital Area Mission Doses (Ref. 3)"
H-1-ZZ-MDC-2021, Rev 2-A, "Salem CR Habitability Due To Post-LOCA Releases From Hope Creek Plant (Ref. 4)"

These analyses were revised to evaluate the radiological impact of additional post-LOCA CST bypass leakage release path. The additional CST bypass leakage path from HPCI & RCIC systems to CST is a new leakage path that was not previously identified and also increases the release of dose consequences in the above four calculations. Therefore, the proposed activity involves a change that adversely affects an UFSAR described design.

The 50.59 evaluation concluded that there was not more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR. The calculations determined that the dose at the Exclusion Area Boundary (EAB) would increase by 0.01 rem, from 3.02 rem to 3.03 rem. The regulatory limit at the EAB is 25 rem. This increase is considered to be a minimal increase in dose consequences. Dose impact to other areas was negligible. Based on this, the evaluation concluded that there would be no increase in the probability or consequences of an accident or malfunction, and each of the eight evaluation questions was answered no.