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LR-N03-0173
Bases Change H03-04B

United States Nuclear Regulatory Commission
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Washington, DC 20555

TECHNICAL SPECIFICATION BASES CHANGE
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

PSEG Nuclear LLC is providing a revised Technical Specification (TS) Bases page for TS 3/4.4.6. The revised page was reviewed in accordance with the requirements of 10 CFR 50.59.

TS 3/4.4.6 pertains to reactor coolant system pressure and temperature limits. The TS Bases have been revised to state that the maximum specified heatup and cooldown rates apply to reactor coolant temperature.

Attachment 1 contains the revised page for the Hope Creek Technical Specification Bases. Please incorporate these changes into the Technical Specification Bases.

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

Sincerely,

G. Salamon
Manager – Nuclear Safety & Licensing

Attachment

A001

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Attachment 1

LR-N03-0173
BASES CHANGE H03-04B

HOPE CREEK GENERATING STATION
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REVISIONS TO THE TECHNICAL SPECIFICATIONS BASES

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (3.9) of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Specifically the average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any 1-hour period.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G and ASME Code Cases N-588 and N-640. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.