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MILLER,C.L. Project Directorate I-2

SUBJECT: Transmits changes to TS Bases 3/4.4.6, "Pressure/Temp Limits," per NRC concurrence w/util design bases review conclusion re vessel heatup/cooldown rates,in ref to suppl info to EA 89-042,dtd 891221.

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JUN 13 1994

Director of Nuclear Reactor Regulation
Attention: Mr. C. L. Miller, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
CHANGES TO THE SSES
TECHNICAL SPECIFICATION BASES FOR
TECHNICAL SPECIFICATION 3/4.4.6
PLA-4144 FILES A17-2/R41-2

Docket Nos. 50-387
and 50-388

Reference - *PLA-3235, H.W. Keiser to W.F. Kane, "Supplemental Information To Enforcement Action 89-042," dated December 21, 1989.*

Dear Mr. Miller:

The purpose of this letter is to transmit changes to the Bases of the Susquehanna Unit 1 and Unit 2 Technical Specifications. The change revises the existing Technical Specification Bases 3/4.4.6, "Pressure/Temperature Limits".

BACKGROUND

In January of 1989, Susquehanna Unit 1 experienced an apparent cooldown rate that exceeded the limits of Technical Specification 3/4.4.6. PP&L's engineering analysis concluded that there were no safety significant issues associated with the incident. As a follow-up, PP&L initiated a review of the design bases surrounding Technical Specification 3/4.4.6 and submitted the findings to NRC (PLA-3235).

PP&L's design bases review concluded that vessel heatup/cooldown rates as required by Technical Specification 3/4.4.6 should be monitored using saturation temperature (T_{SAT}). The NRC concurred with this conclusion and PP&L committed to a revision of the Bases section of Technical Specification 3/4.4.6 as final close-out of this issue.

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DESCRIPTION OF CHANGE

The attached change to Technical Specification Bases section 3/4.4.6 reflects a complete rewrite of this section that documents that an adequate margin of safety against brittle fracture of the reactor pressure vessel (RPV) is maintained.

SAFETY ANALYSIS

The principal design criteria for the Reactor Coolant Pressure Boundary (RCPB) requires that it retain integrity as a radioactive material barrier during normal operation following abnormal operational transients and accidents. Compliance with this criteria is assured by satisfying requirements of paragraph 50.55a and the General Design Criteria of Appendix A of 10CFR50.

The General Design Criteria require that the RCPB possess adequate toughness properties to resist brittle fracture during all conditions of plant operation. These fracture toughness requirements are satisfied by complying with Appendix G of 10CFR50. Compliance is achieved by assuring that materials in the RCPB meet specified toughness requirements and that specific pressure/temperature (P/T) operating limits are applied to the reactor vessel. Additionally, a reactor vessel material surveillance program is required to monitor changes in fracture toughness of the vessel beltline materials due to neutron embrittlement. These measures assure a wide margin to brittle fracture of the RCPB thus ensuring its structural integrity as required by the principal design criteria.

Prevention of brittle fracture for the RCPB is assured by conformance to General Design Criterion 31. This criterion applies particularly to the RPV since it is the component of the RCPB which is most susceptible to brittle fracture.

Design conformance with Criterion 31 is demonstrated by requiring that the RPV meet the requirements of Appendix G of 10CFR50. Appendix G requires that the RPV be designed and analyzed to meet the requirements of Appendix G of Section III of the ASME Code. This Code appendix provides an analysis procedure for deriving the RPV P/T operating limits. The analysis is designed such that the derived operating limits provide a wide margin of safety to brittle fracture. The P/T limit curves which are contained in Technical Specification 3/4.4.6 are the results of the Appendix G analysis.

All analysis results and requirements have been incorporated into the limits of Technical Specification 3/4.4.6. Since the potential for violating any of these limits exists any time the reactor can be pressurized or a vessel temperature gradient is present, the limits are applicable at all times. If the limits are exceeded the appropriate action is to restore the vessel to within limits since this puts the vessel back into its analyzed envelope for brittle fracture. Additionally, an analysis is required for such violations to assure that RCPB integrity is acceptable for continued operation.

The transient design bases for the vessel assumes a normal operating heatup/cooldown rate of 100°F/hr. Given the model from which the thru-wall temperatures of the vessel are computed, the 100°F/hr rate during heating or cooling which is used for brittle fracture analysis purposes refers to an instantaneous rate. However, per GE Report SASR 89-40, "Pressure-Temperature Curve Basis for Susquehanna SES Units 1 & 2, June 1989, instantaneous rates in excess of 100°F/hr are allowed for in the Technical Specification as long as a temperature range of 100°F in a one hour period is not exceeded. However, it is understood in this Technical Specification allowance that operators will track vessel coolant heatup or cooldown to stay as close to a 100°F/hr rate as possible.

In summary, PP&L's investigation of the relevant design bases concluded that brittle fracture mitigation for the reactor vessel is the intent of Technical Specification 3/4.4.6. Fracture prevention and fracture toughness requirements are prescribed by 10CFR50 Appendix G. The P/T curves in this Technical Specification were derived using linear elastic fracture mechanics techniques in accordance with ASME Section III Appendix G. Analysis for the beltline and non-beltline regions were performed. The Technical Specification P/T curves are composite worst case bounding results of this fracture analysis and provide a wide margin of safety to brittle fracture of the reactor vessel.

The fracture analysis performed considers the effects of neutron embrittlement of the beltline region. 10CFR50 Appendix H requires that a program be in place which monitors actual embrittlement effects throughout the life of the plant. Susquehanna demonstrates compliance with this requirement via Technical Specification surveillances. Proper vessel monitoring is required in order to demonstrate operational compliance with Technical Specification 3/4.4.6 limits.

All results of PP&L's analyses have been incorporated into the revision to the Bases section of Technical Specification 3/4.4.6. PP&L's review and approval of this document assures that the revision has not compromised any intended safety margins imposed by this Technical Specification.

IMPLEMENTATION

PP&L has implemented this change based on a review by the Plant Operations Review Committee. The attached revised pages are provided for your information and use.

Questions regarding the above proposal can be directed to Mr. J.B. Wesner at (610) 774-7911.

Very truly yours,

R.G. Byram

Attachments

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector
Mr. C. Poslusny, Jr., NRC Project Manager