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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards proposed Amend 24 to License NPF-14,changing App A Tech Specs,re circuit breaker location,gaseous release type, bypass damper leakage & reactor vessel level instrumentation.

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Direction of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT 24 TO LICENSE NPF-14
ER 100450 FILE 841-8
PLA-1645

Docket No. 50-387

Dear Mr. Schwencer:

The purpose of this letter is to propose changes to Appendix A of License No. NPF-14, the Susquehanna SES Unit 1 Technical Specifications. The proposed changes are provided as attachments to this letter.

ITEM 1: Table 4.11.2.1.2-1
PROPOSED CHANGE: Revise per Attachment A.
JUSTIFICATION: Both of these changes serve to correct typographical errors in the table. The asterisk footnote on "Gaseous Release Type" has no associated words, and footnote "g" not "b" applies to "Principal Gamma Emitters." Each of these items was checked against the Standard Technical Specifications, and the proposed changes make the table consistent with this standard. Neither of these changes impact the safe operation of Susquehanna SES.

ITEM 2: Various references to systems or equipment in Section 8.0, "Electrical Power Systems."
PROPOSED CHANGE: Revise per Attachment B (8 pages).
JUSTIFICATION: Except for two items, each of the proposed changes is a typographical correction. The first is in Table 4.8.1.1.2-2. The tag numbers on these two devices have been changed in the field and the Tech Specs are being updated to reflect that change. The second item is the footnote being added to Table 3.8.4.1-1 on part a, "Type 2 Molded Case Circuit Breakers." The purpose of this change is to clarify the fact that each number under "Circuit Breaker Location" represents two breakers in series, both of which are to be tested.

Appropriate plant records have been reviewed to insure that no surveillances were missed due to any of these proposed changes. This review proved to be satisfactory and this change therefore has no impact on the safe operation of Susquehanna SES.

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ITEM 3: Specification 4.7.2.b.1
PROPOSED CHANGE: Revise per Attachment C (2 pages).
JUSTIFICATION: The purpose of this change is to delete a requirement that does not reflect the as-built design of Susquehanna SES Unit 1. Specification 4.7.2.b.1 is a carry-over from the BWR 4 Standard Tech Specs which should have been recognized in earlier reviews as a non-applicable item.

The purpose of the surveillance requirement is to control and minimize leakage through the bypass damper, if provided as part of the CREOASS design. The design of Susquehanna does not and was never intended to include a bypass connection.

The proposed change has no impact on the safe operation of Susquehanna SES, as the absence of this leakage pathway creates a more conservative configuration.

ITEM 4: Table 4.4.6.1.3-1
PROPOSED CHANGE: Revise per Attachment D.
JUSTIFICATION: The proposed change corrects an inconsistency between the Tech Specs and 10CFR50 Appendix H. As described in FSAR subsection 5.3.1.6.1, the withdrawal schedule associated with the reactor vessel material surveillance program per 10CFR50 Appendix H will be as follows:

a. The first set will be withdrawn at 25% of the reactor design life.
b. The second set will be withdrawn at 75% of the reactor design life.
c. The third set will be a standby.

The design life of the Susquehanna SES Unit 1 reactor is 32 Effective Full Power Years (EFPY). The proposed change reflects the resulting schedule. The change has no impact on the safe operation of SSES; it simply alters the schedule to make it consistent with the existing regulations.

ITEM 5: Table 3.3.9-2
PROPOSED CHANGE: Revise per Attachment E
JUSTIFICATION: The Reactor Vessel Level-High Feedwater/Main Turbine trip instrumentation is provided to protect against the occurrence of gross carryover of moisture to the main turbine and to prevent vessel overfill by the feedwater pumps. These two functions are not safety related and are not considered part of the safety analysis in the FSAR. However, the transients analyzed in Chapter 15 of the FSAR assume normal operation of the High Reactor Vessel Level Feedwater/Main Turbine trip instrumentation. Therefore, the effect on the transient analyses in Chapter 15 must be assessed prior to changing the High Reactor Vessel Level Feedwater/Main Turbine trip Allowable Value in the Technical Specifications.

General Electric's recommended settings are given below:

Reactor Vessel Level - High

a.	Setpoint	54.0 inches
b.	Allowable Value	55.5 inches
c.	Analytic Limit	58.7 inches

The differences between the three values above reflect the uncertainties in the instrumentation and calibration. A review of the FSAR Chapter 15 disclosed that the value for the Reactor Vessel Level - High trip used in the transient analysis was 54.0 inches which does not support the GE recommended values. Therefore, the NRC set the SSES Technical Specification Allowable Value to the value used in the transient analysis.

At PP&L's request, GE performed an analysis of the Feedwater controller failure - maximum demand transient using a Reactor Vessel Level - High trip of 58.7 inches to support the GE recommended Technical Specification values. The results of the analysis indicate that the Minimum Critical Power Ratio (MCPR) operating limit remains the same as currently analyzed in the FSAR. All other events in Chapter 15 are unaffected by an increase in the High Reactor Vessel Level trip value since either: (1) the Reactor Vessel Level - High trip is not initiated during the transient, or (2) no power spike occurs as a result of the High Vessel Level turbine trip due to decreasing power conditions prior to the trip. This Technical Specification change does not impact the overpressurization analysis (FSAR Chapter 5.2) due to the same phenomenon as discussed in (2) above. In addition, the LOCA analysis (FSAR Chapter 6.3) results in decreasing water level transients which are not affected by the High Reactor Vessel Level trip. The FSAR will be updated, upon NRC approval of this Technical Specification change, to include the results of the GE feedwater controller failure - maximum demand analysis with a 58.7 inch High Vessel Level trip. Therefore, the effects of increasing the Reactor Vessel Level - High Feedwater/Main Turbine trip Allowable Value to 55.5 inches have been analyzed be GE. The results indicate that no degradation to plant safety results from this Technical Specification change.

SIGNIFICANT HAZARDS CONSIDERATIONS

PP&L has reviewed each of the items above and has made the following determinations with respect to the criteria listed in 10CFR50.92(c):

ITEM 1: No significant hazard. This item is purely administrative.

ITEM 2: No significant hazard. This item is purely administrative.

ITEM 3: No significant hazard. This item is administrative, in that the staff has already reviewed and accepted the as-built CREOASS design, which includes no bypass connection.

ITEM 4: No significant hazard. This item is administrative, in that the withdrawal schedule is simply being revised to be consistent with existing FSAR commitments and federal regulations.

ITEM 5: No significant hazard. This change is being proposed in order to make the FSAR analysis consistent with GE's design recommendations. As described above, the analysis performed resulted in no significant reduction in the margin of safety, as measured by the fact that the MCPR Operating Limit remained unchanged. The results of the analysis also show that this change:

1) involves neither a significant increase in the probability of an accident previously evaluated, nor
2) creates the possibility of a new or different kind of accident from any accident previously evaluated.

We have determined items 1 through 4 to be a Class II change and item 5 to be a Class III change pursuant to 10CFR170.22 and have enclosed the appropriate fees.

Very truly yours,

B. D. Kenyon
Vice President-Nuclear Operations

Attachments

cc: R. L. Perch - USNRC