DRAFT EVALUATION OF TASK INTERFACE AGREEMENT 2000-10

FARLEY NUCLEAR PLANT, UNITS 1 AND 2

REACTOR VESSEL SUPPORT CONCRETE TEMPERATURE

INTRODUCTION

Region II issued non-cited violation (NCV) 50-348, 364/00-01-01 to Southern Nuclear Operating Company (SNC) based on NRR's January 27, 2000, response to Task Interface Agreement (TIA) 98-11, "Farley's Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures." The NCV stated that sustained reactor vessel support (RVS) concrete temperatures above 150°F constitute an unreviewed safety question (USQ). SNC's letter of May 31, 2000, denied the violation and gave us more information to support their denial. Region II's TIA 2000-10 of June 23, 2000, asked NRR to review the additional information and confirm or modify our earlier TIA response as appropriate. This evaluation responds to Region II's request of June 23, 2000.

EVALUATION

Attachment 3 to SNC's letter of May 31, 2000, contained SNC's rebuttal to NRR's response to TIA 98-11. The staff reviewed SNC's rebuttal to the three questions addressed in Attachment 3 to SNC's letter as shown below.

A. SNC's Response to Question 1

This question relates to the appropriateness of using the American Concrete Institute (ACI) code limit of 200°F for RVS concrete. In its response to TIA 98-11 (Ref. 2), the staff stated that the 200°F limit should be applied to the localized concrete areas around the high-energy pipes passing through a concrete structure. It should not be applied to principal load bearing concrete components such as RVS concrete without proper technical justification.

In the first two paragraphs of SNC's response to question 1, SNC attempted to interpret ACI 349 code limits to support its argument that the ACI 349 code limit of 200°F apples to RVS concrete. SNC provided additional information on RVS structure attributes in subsequent paragraphs.

In its response to TIA 98-11, the staff considered that the RVS structure concrete was subjected to sustained temperatures above 190°F. SNC pointed out that 2 out of the 6 supports are at temperatures below 150°F, 2 are at about 165°F, and the remaining 2 are at about 190°F. This information does support SNC's argument about the localized nature of the temperatures above 150°F. However, it raises a question about differential settlement of the supports, and potential effects on the supported nozzles and the reactor vessel. This is one aspect of the USQ.

Attachment

Additionally, SNC points out that the RVS structures are welded to the reactor cavity wall liner plate which would transfer heat into the concrete, thus potentially reducing the local peak calculated concrete temperature. Sheet 1 of Attachment 4 shows the liner plate is ¼-inch thick. The liner plate does somewhat confine the concrete. However, the liner may have high compressive strains and potential bulging due to varying thermal gradients through the primary shield wall below the RVS structure. This depends on how the liner is anchored to the primary shield wall below the RVS structure. Thus, the existence of the liner may provide some relief in the calculated concrete temperature, but gives rise to an additional structural issue under high-temperature gradients. SNC should address this issue as part of the USQ.

B. SNC's Response to Question 2

This question relates to the appropriateness of SNC applying the ACI code limits and whether continually exceeding these limits constitutes a USQ. In its earlier response, the NRC staff determined that SNC should have identified the RVS support condition as a USQ in its 10 CFR 50.59 evaluation. This is because exceeding the temperature limits could result in an increase in the probability of malfunction of equipment important to safety (i.e. the RVS) as previously evaluated in the UFSAR.

SNC points out that the temperatures of RVS structures remain between 120°F and 190°F, contrary to the staff's estimates that they will remain above 190°F. The implication of this difference is discussed in item 1. above. However, SNC's response does not address the staff's basic concern related to the USQ (i.e., potential malfunction of the RVS concrete).

C. SNC's Response to Question 3

This question is related to the potential safety consequence of exceeding the ACI code limits of 150°F or 200°F for the RVS concrete. The staff's response to TIA 98-11 provided the results of available research related to the properties of concrete at these temperatures.

SNC's response provided only a qualitative assessment of the potential actual strength of the primary shield wall and RVS concrete. The staff considers it justifiable to use statistically reliable actual concrete strength data compared to specified design strength. The associated increases in modulus of elasticity and Poisson's ratio could then be determined. The licensee could demonstrate that in spite of the increased RVS concrete temperatures (and corresponding decrease in the concrete properties) the associated stresses do not exceed design-basis acceptance criteria under postulated loadings. Such an analysis, if found acceptable, could resolve the USQ.

CONCLUSION

The Farley plant was not designed to the ACI 349 code. However, Westinghouse's 1973 generic thermal analysis of the RVS structure (Ref. 3) stated that the maximum temperature of the bottom surface of the RVS structure in contact with the concrete must be maintained at $\leq$ 150°F.

In response to SNC's question regarding the Farley plant, in a letter of October, 28, 1998 (Ref. 4), Westinghouse stated that the temperature limit mentioned in its 1973 report was the generic limit that Westinghouse used for all plant designs and was based on the

recommendations for general-area temperature limits. Westinghouse further stated that the customer or Architect Engineer, if necessary, could extend this limit. However, plant-specific technical justification might be necessary for this extension.

The staff's view is that the code requirements may not be succinct enough to cover all the possible scenarios. The issue of significance here is not the strict interpretation of code requirements, but the safety concern associated with the sustained high temperatures. For example, the staff has accepted temperatures higher than 150°F in spent fuel pool (SFP) concrete for short-term transients that could occur during the full core load in the pool, and assuming the unavailability of the make-up water. The staff judged that such occasional temperature transients, if they were to occur, would not affect the properties of the SFP concrete.

Based on its review of SNC's reply on May 31, 2000, the staff concludes that SNC has not given us any data or analysis that minimizes the staff's concern about the potential malfunction of the RVS concrete or alters the staff's determination that a USQ exists.

<u>REFERENCES</u>

1. Letter, Dave Morey, SNC, to NRC, "Reply to NCV 50-348, 364/00-01-01, Failure to Identify an USQ," May 31, 2000.

2. Staff evaluation of Task Interface Agreement 98-11, "Farley's Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures," January 27, 2000.

3. ED-THA-7: Westinghouse Report, "RVS Structure Thermal Analysis - Parametric Study," October 22, 1973 (Attachment 3 to Region II's TIA 98-11 of December 8, 1998).

4. Letter, Westinghouse to SNC, "Reactor Vessel Concrete Support Temperatures," October 28, 1998 (Attachment 4 to Region II's TIA 98-11 of December 8, 1998).

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