March 6, 2001

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects
Region II

FROM: Suzanne C. Black, Deputy Director ***/RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT 2000-10 RE: EVALUATION OF TIA 98-11 REGARDING REACTOR VESSEL SUPPORT CONCRETE TEMPERATURE AT FARLEY (TAC NOS. MA9311 AND MA9312)

Region II's Task Interface Agreement (TIA) 2000-10 of June 23, 2000, requested our assistance to review Southern Nuclear Operating Company (SNC) additional information on a reactor vessel support (RVS) concrete temperature Unreviewed Safety Question (USQ) at Farley. TIA 2000-10 requested that we either re-confirm or modify our earlier TIA 98-11 response that the issue at Farley was a USQ.

The attached evaluation contains our response to your question. SNC's letter of November 10, 2000, said that SNC (1) inspects the outside accessible surfaces of the primary shield wall concrete as part of its structural monitoring program, and (2) assesses the reactor vessel plumbness based on observations during refueling activities that would provide indications regarding the plumbness of reactor vessel. These actions are part of Farley's maintenance and operational activities. The staff considers that with the stated focus on RVS degradation, SNC's current practice provides reasonable assurance that SNC will identify the effects of sustained high temperatures on the vessel supports and will take any necessary remedial actions. Thus, the NRR staff concludes that SNC's current practice resolves this issue. We discussed our proposed response with Steve Cahill in February 2001.

SNC added the following words to Farley plant UFSAR Section 5.5.14.1.A to document their augmented program to inspect the structural components including portions of the reactor vessel support:

"However, recognizing the potential degradation of the RPV supports subjected to sustained temperatures higher than 150°F, FNP has committed (NEL letter #00-279 to USNRC) to an augmented program to inspect the structural components including portions of the RVS in the containment buildings as part of the maintenance rule structural monitoring program. This program will ensure that significant cracking of RVS that could affect the structural support of the reactor vessel or cause out of plumbness conditions will be detected and corrected."

Docket Nos. 50-348 and 50-364

Attachment: As stated

cc w/att: M. E. Oprendek, Region I
G. E. Grant, Region III
K. E. Brockman, Region IV

CONTACT: Mark Padovan, NRR/DLPM
(301) 415-1423

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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 (Ref. 1), "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 (Ref. 2), denied the violation and provided more information to support its denial. Region II's TIA 2000-10 of June 23, 2000, requested NRR to review the information and confirm or modify its earlier TIA response as appropriate.

Subsequently, NRC staff held a phone conversation with SNC on September 19, 2000, on how to possibly resolve the issue. The staff proposed two actions that SNC could take to monitor potential consequences of the sustained high temperatures around some of the reactor pressure vessel (RPV) support concrete areas. The staff requested SNC to focus their inspections on identifying potential degradation on the outside surfaces of the primary shield wall concrete. The staff also requested SNC to monitor the plumbness of the reactor vessel during operational and maintenance activities.

SNC's letter of November 10, 2000 (Ref. 3), said that SNC will (1) inspect outside accessible surfaces of primary shield wall concrete as part of its structural monitoring program, and (2) assess reactor vessel plumbness as part of the activities that would indicate reactor vessel plumbness (i.e., abnormal loading or handling difficulties during reactor head and reactor internals disassembly). SNC stated that these actions were part of its maintenance and operational activities.

The following evaluation discusses the additional information SNC provided in References 2 and 3.

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.

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. 1), 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 applies to RVS concrete. 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 ≤ 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.

SNC provided additional information on RVS structure attributes in subsequent paragraphs of its response. 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. Reference 3 shows that SNC is monitoring the plumbness of the reactor vessel. Thus, if differential settlement of the RPV support occurs, it will result in an out-of-plumb condition of the reactor vessel, and SNC will take appropriate remedial actions. Thus, SNC has addressed the staff's concern related to the differential settlement of the RPV supports.

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 (Ref. 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. As shown in Reference 3, SNC is monitoring the condition of RPV supports as part of its structural monitoring program. Thus, SNC is monitoring any undesirable deformation of the liner. This addresses the staff's concern related to the liner bulging.

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 TIA 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 A. above. Reference 3 shows that SNC is monitoring the plumbness of the reactor vessel as part of its operational activities. The staff believes that SNC is monitoring any malfunction resulting from the RVS concrete degradation and will take any necessary remedial actions. Thus, no USQ exists.

C. SNC's Response to Question 3

This question is related to the potential safety consequences 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 a qualitative assessment of the potential actual strength of the primary shield wall and RVS concrete. The technical concern raised in this question is no longer relevant as a result of the licensee's actions discussed in the introduction of this evaluation.

CONCLUSION

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. In Reference 3, SNC states that activities are in place to adequately monitor the condition of the RVS concrete and to detect significant reactor vessel out-of-plumb conditions. The staff believes that these monitoring actions will detect the manifestation of concrete degradation, as well as any resulting out-of-plumb condition of the reactor vessel at Farley. Thus, the staff considers that SNC's existing monitoring actions adequately address this issue and no USQ exists.

REFERENCES

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

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

3. Letter, Dave Morey, SNC, to NRC, "Additional Information to NCV 50-348, 364/00-01-01, Failure to identify a USQ," November 10, 2000.

4. 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).

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