January 27, 2000

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 98-11, FARLEY'S INTERPRETATION OF ACI CODE FOR REACTOR VESSEL SUPPORT CONCRETE TEMPERATURES (TAC NOS. MA4397 AND MA4398)

Region II's Task Interface Agreement (TIA) 98-11 of December 8, 1998, requests our assistance to answer the following questions about the Farley Nuclear Power Plant, Units 1 and 2, reactor vessel supports (RVS):

1. Is the licensee [Southern Nuclear Operating Company] correct in applying the ACI [American Concrete Institute] code limit of 200 degrees Fahrenheit for localized areas of the reactor vessel supports?

2. If the licensee is improperly applying the code limit or exceeding 200°F for the reactor vessel supports, does an Unreviewed Safety Question [USQ] exist?

3. What is the actual or potential safety consequence for exceeding the ACI code limit of 150°F or 200°F for the RVS concrete?

4. Is the licensee's analysis of RVS concrete temperature adequate, i.e., could they be exceeding 200°F at the concrete?

The attached evaluation contains our responses to these questions. We discussed our proposed responses with Pierce Skinner and Jon Bartley in September, October, November, December 1999, and January 2000. Please contact Mark Padovan at (301) 415-1423 if you have any questions.

Docket Nos. 50-348 and 50-364

Attachments: As stated

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

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**J. Bartley & P. Skinner of RII agreed to the release TIA to the public. P. Tam 12/22/99

EVALUATION OF TASK INTERFACE AGREEMENT 98-11

FARLEY NUCLEAR PLANT, UNITS 1 AND 2

INTERPRETATION OF ACI CODE

FOR REACTOR VESSEL SUPPORT CONCRETE TEMPERATURES

INTRODUCTION

By memorandum, L. Plisco to J. Zwolinski, dated December 8, 1998, Region II submitted Task Interface Agreement (TIA) 98-11 to the Office of Nuclear Reactor Regulation (NRR). The following three paragraphs cited verbatim from TIA 98-11 provide background information related to the TIA concerns.

> In 1976, the licensee [Southern Nuclear Operating Company] identified that the reactor cavity cooling system, which cools the Reactor Vessel Supports (RVS)(Att.1), was not performing as designed. The licensee performed a calculation and determined that the system performance was acceptable. This conclusion was based on the licensee's interpretation of an American Concrete Institute (ACI) code which allowed concrete temperatures up to 200 degrees Fahrenheit (°F) in localized areas. At that time, a change to section 5.5.14.1.A of the Final Safety Analysis Report (FSAR) adopting the new localized temperature value was not identified. This missed change to the FSAR was not recognized until the FSAR verification effort in 1996.
>
> In 1997, the licensee approved a change to the FSAR to increase the allowable reactor vessel concrete support temperature from 130 °F to 190 °F. The resident inspector's review of the licensee's change indicated that this change may have violated the requirements of 10 CFR 50.59 in that an unreviewed safety question may exist. This issue was documented in Inspection Report 50-348, 364/98-05 as EEI 50-348, 364/98-05-02.
>
> There have been numerous discussions between the NRC and licensee staff on this issue. The original NRC staff concern was about the interpretation of the ACI code concerning local area temperatures. However, the inspector's follow up review has identified that the concrete temperatures may also exceed the local area temperature code limit. The licensee is planning to visually inspect several supports during the reactor vessel nozzle inservice test. However, based on construction photographs, it appears that very little of the RVS concrete will be visible. The licensee does not plan to perform any further calculations or analyses.

This evaluation responds to the specific questions identified by Region II in TIA 98-11.

EVALUATION

The reactor pressure vessel (RPV) at each unit of the Farley Nuclear Plant (FNP) is supported by six nozzles (three hot-leg and three cold-leg). The nozzles, in turn, are supported by the RVS. The RVS are steel box structures beneath each vessel nozzle that are secured to the primary shield wall.

The reactor cavity cooling system (RCCS) consists of fans, dampers, and ductwork, and is not a safety-related system. A function of the system is to provide forced convection cooling flow through the six RVS. Balancing dampers in the duct ensure proper flow distribution from the individual RVS. In a licensee event report (LER) 98-08-01 related to the closed cooling damper (Ref. 1) SNC states that the RCCS is designed to maintain the concrete at the RVS within the guidance provided by Article CC-3440(a) of Section III, Division 2 of the ASME Code. SNC applies the temperature limit of 200 °F to the concrete in contact with the RVS.

Our responses to Region II's questions are as follows:

Q1 *Is the licensee correct in applying the ACI code limit of 200 degrees Fahrenheit for localized areas of the reactor vessel supports?*

R1 No. The concrete under the RVS is subjected to significant loadings caused by the dead load of the RPV and lateral loads due to transients and seismic loads. The staff's understanding of the 200 °F code limit is that it applies to some localized areas within a structure, but should not be applied to the principal load-bearing concrete, such as the concrete bearing the RVS loads.

Q2 *If the licensee is improperly applying the code limit or exceeding 200°F for the reactor vessel supports, does an Unreviewed Safety Question [USQ] exist?*

R2 Yes. When SNC determined that the actual concrete temperatures near the RVS were above 130 °F [as stated in the current Updated Final Safety Analysis Report (UFSAR)], SNC was required to evaluate the issue in accordance with 10 CFR 50.59. Based on the resident inspector's analysis (Attachment 7 to TIA 98-11), and LER 98-08-01 (Ref. 1), the staff concludes that the concrete temperatures in the vicinity of the RVS are and will remain above 190 °F. It is the staff's view that such a high temperature should have been designated a USQ, because it could result in the probability of malfunction of equipment important to safety (i.e., the RVS), as previously evaluated in the UFSAR, to increase.

Q3 *What is the actual or potential safety consequence for exceeding the ACI code limit of 150°F or 200°F for the RVS concrete?*

R3 Available information (Ref. 2) indicates that sustained temperatures up to about 150 °F cause insignificant changes to concrete properties (i.e., compressive strength, modulus of elasticity, Poisson's ratio). At about 190 °F, the reduction in compressive strength is about 10 percent, the reduction in the modulus of elasticity is about 30 percent, and the reduction in Poisson's ratio is about 22 percent. Also, the increase in the compressive strength with time (which is typical at 70 °F) reduces at sustained (> 200 days) high temperatures; and after about 150 °F, the compressive strength starts decreasing with time. The ACI code limits are based on this type of research data.

The changes in the mechanical properties as indicated above are due to the gradual loss of free and chemically bound water in the concrete, which, in turn, leads to a reduction in the concrete stiffness and strength. It should be noted that the above-cited temperature effects were based on testing of non-degraded concrete specimens. In addition to thermal effects, the high flux neutrons and gamma radiation (prevalent around the RVS) adversely affect the physical properties of the concrete. NUREG-1557 (Ref. 3) establishes their threshold levels at $5x10^{19}n/cm^2$ and 10^{10} rads, respectively.

Q4 Is the licensee's analysis of RVS concrete temperature adequate, i.e., could they be exceeding 200 °F at the concrete?

R4 We conclude that SNC's analysis of RVS concrete temperature is adequate and that the peak concrete temperature will not exceed 200 °F for air flow rates ranging between 2000 cfm and 3000 cfm. We evaluated the following three parameters to reach our conclusion:

- RVS air flow rate
- RVS air inlet temperature
- reactor vessel nozzle temperature

We discuss these items in the Evaluation section below.

I. Background

The RVSs are steel and concrete box structures beneath each reactor vessel nozzle that are secured to the primary shield wall. Hot reactor vessel nozzles heat the RVSs, so the reactor cavity cooling fan pulls air through the RVSs to cool the structures.

SNC's thermal analysis models the RVSs and the surrounding concrete at the bottom of the RVS. SNC's analysis consists of two separate analyses. The primary analysis is Farley-specific and is documented in a report titled, "Reactor Vessel Support Thermal and Thermal Stress Analysis," which is an Appendix to Westinghouse letter PA-MSA-489 (Ref. 4).

The secondary analysis consists of a series of parametric analyses. These analyses are documented in a report EO-THA-7, "RVS Structure Thermal Analysis - Parametric Study." The thermal modeling in these analyses is for a typical RVS design which does differ somewhat from the Farley design. However, there are enough similarities to determine parameter trends (increases or decreases) as opposed to absolute values. Determining parameter trends as opposed to absolute values is acceptable, but the results of these analyses are to be considered generic rather than Farley-specific.

One note of caution should be made. Our evaluations are based on best-estimate assumptions for the most part. Only where it could not be explicitly determined were conservative assumptions or calculations applied in the evaluation. Without any further guidance, the results should be considered to be best-estimate.

II. Evaluation

Our evaluation addresses the effects of RVS air flow rate, RVS air inlet temperature, and reactor vessel nozzle temperature on RVS concrete as shown below.

Different Design Assumptions

The Farley plant-specific analysis showed the RVS bottom concrete temperature would be 155 ^oF at design conditions. The assumed design conditions are as follows:

- 3000 cfm RVS air flow rate
- 120 ^oF RVS air inlet temperature
- 547 ^oF reactor vessel nozzle temperature

Two of the design assumptions in our evaluation are different from SNC's. SNC had test data which showed that the RVS air flow rate could be as low as 2000 cfm rather than the design level of 3000 cfm. Also, reactor vessel nozzle temperature depends upon whether the support is for a hot leg or a cold leg. At 100 % power, the hot leg is expected to be at 613 ^oF. We assessed the effects of these different design assumptions and the 120 ^oF RVS air inlet temperature on RVS bottom concrete temperature below.

RVS Air Flow Rate

SNC's test data showed that RVS air flow rate could be as low as 2000 cfm. Figure 7 in EO-THA-7 shows that the bottom RVS temperature will increase by 18 ^oF when the flow rate is reduced from 3000 cfm to 2000 cfm. SNC performed a similar evaluation using the same analyses. Their evaluation results are documented in Westinghouse's ALA-98-261 letter to Southern Nuclear Operating Company dated October 28, 1998. Westinghouse looked at Figure 7 and concluded that the temperature would increase between 15 ^oF and 20 ^oF. This agrees with our assessment that the bottom RVS temperature will increase by 18 ^oF. Therefore, the effect of the reduced air flow would yield concrete temperatures ranging between 170 ^oF and 175 ^oF (155 ^oF from the plant-specific analysis + 15 ^oF to 20 ^oF). SNC then added a final correction factor of 15 ^oF to the conservative 175 ^oF figure to reach 190 ^oF. SNC added the 15 ^oF correction factor because the Farley RVS design did not include cooling fins that were accounted for in the typical RVS design. Thus, peak RVS concrete temperature will not exceed 200 ^oF for air flow rates ranging between 2000 cfm and 3000 cfm.

RVS Air Inlet Temperature

The Farley plant-specific analysis used a 120 ^oF RVS air inlet temperature design condition. This is acceptable since our evaluation shows a near-uniform air temperature of 123 ^oF within the air passages. Our 123 ^oF figure is based on the following:

- Figure 5 in EO-THA-7 indicates that RVS air outlet temperature is 123 ^{o}F.
- The air is almost isothermal for the expected conditions (we explain this below).
- Air cavity surface temperatures are more or less uniform (we also explain this below).

We looked at Table 2 in EO-THA-7 to determine that the RVS air is almost isothermal for the expected conditions. Table 2 gives air temperature rise as a function of both air inlet temperature and air flow rate. There is only a small increase in air temperature from inlet to outlet. For example, assuming an inlet air temperature of 120 ^{o}F, the air temperature rise is only 4.6 ^{o}F at a flow rate of 3000 cfm. For an air flow rate of 2000 cfm, the air temperature rise is only 6.60 ^{o}F. This means that the air is almost isothermal for the expected conditions.

We used Figure 5 in EO-THA-7 to determine that surface temperatures of the RVS air cavities are more or less uniform. Figure 5 shows that surface temperatures varied between 127 ^{o}F and 146 ^{o}F. Most importantly, Figure 5 shows these surface temperatures are for a reactor coolant system (RCS) (i.e., reactor vessel nozzle) temperature of 613 ^{o}F. However, it should be noted that Figure 5 also assumes air flow rate of 4800 cfm. However, based on the previous discussion, the air flow rate does not significantly affect the RVS temperatures.

<u>Reactor Vessel Nozzle Temperature</u>

In response to an NRC staff question, SNC assessed the effect of increasing the RCS temperature from 547 ^{o}F to 613 ^{o}F on RVS concrete temperature. SNC responded that RVS concrete temperature would increase by less than 2.5 ^{o}F. SNC based their conclusion on a 1973 analysis that showed that RVS concrete temperature increased less than a 0.5 ^{o}F when RCS temperature increased from 605 ^{o}F to 613 ^{o}F.

Our assessment concluded that RVS concrete temperature increased 1 ^{o}F when RCS temperature increased from 605 ^{o}F to 613 ^{o}F. This RVS concrete temperature increase is insignificant and does not need to be considered in RVS analyses. Our bases for this follows.

We took air inlet temperature and air flow rate from Figure 5 in report EO-THA-7 and used Figure 7 to determine a maximum concrete bottom temperature of 147 ^{o}F. Therefore, the increase in RCS temperature caused a 1 ^{o}F increase in RVS concrete bottom temperature which is an insignificant effect.

Also, SNC's Farley plant-specific analysis contains additional conservatism as can be seen from several figures and graphs showing thermal gradients from the RCS piping to the bottom of the RVS concrete. The thermal profiles show an important thermal characteristic. Beginning at a RCS temperature of 547 ^{o}F, the temperature drops to 300 ^{o}F within a very short distance from the RCS. As a result, the bulk of the surrounding surfaces that are air-cooled via the internal air flow path never exceeds 300 ^{o}F. In fact, the bottom concrete surface exposed to the air flow may be limited to 150 ^{o}F. This behavior is indicative of a reasonably well insulated assembly.

An NRC inspector questioned if a change in inlet air temperature would have the same effect on RVS concrete temperature as an equal RCS temperature change. It would not since the thermal contours (temperature patterns) would be different. Calculated concrete temperatures would exceed 200 °F if this were true.

CONCLUSION

Based on information given in TIA 98-11 and its attachments, our evaluation of the first three questions indicates that having sustained temperatures above 190 °F in the RVS concrete constitutes a USQ.

For question four, the staff finds SNC's analyses to be acceptable. This means that we concur with SNC that the peak concrete temperature will not exceed 200 °F for air flow rates ranging between 2000 cfm and 3000 cfm. The analyses that were evaluated considered only conditions where all systems were considered operable. Analyses which considered only natural circulation were not evaluated since they were considered beyond the scope of TIA 98-11.

Principal Contributors: J. A. Kudrick
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3. NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resource Council Industry Reports Addressing License Renewal," U.S. Nuclear Regulatory Commission, October 1996.

4. Westinghouse, letter PA-MSA-489, "Reactor Vessel Support - Thermal and Thermal Stress Analysis Farley Plant," July 20, 1972.