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SOUTHERN COMPANY
Energy to Serve Your World℠

May 31, 2000

Docket Nos.: 50-348
50-364

NEL-00-0141

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Joseph M. Farley Nuclear Plant
Reactor Vessel Support Concrete Temperature
Reply to NCV 50-348, 364/00-01-01, Failure to Identify an USQ

Ladies and Gentlemen:

As requested by your transmittal dated March 23, 2000, this letter responds to non-cited violation (NCV) 50-348, 364/00-01-01, Failure to Identify an USQ. As agreed by Southern Nuclear Operating Company (SNC) and NRC staff, this response is delayed beyond the originally requested 30 days. The delay was to accommodate discussion between the NRC and SNC staffs regarding the violation. A request to delay the required response to June 1, 2000 was submitted by SNC in a letter dated May 5, 2000. By letter dated May 15, 2000 the NRC staff approved the delay.

The staff has based the NCV on an interpretation of the ACI 349 "Code Requirements for Nuclear Safety Related Concrete Structures". The NRC staff maintains that application of a maximum 200°F localized temperature limit to the reactor vessel supports (RVS) is not allowed since the RVS are load bearing structures. After review, SNC has concluded that the 200°F limit is in accordance with the FNP licensing basis, applicable codes and industry practice and therefore, denies the subject NCV. To further support this conclusion, SNC has contacted members of the ACI 349 code committee, including the Subcommittee chairman of design who was involved with incorporating the concrete temperature limitations into the code. These members confirm that the 200°F localized limit was not intended to exclude application to principal load-bearing concrete structures. They also confirm that the Code committee was aware that there were small changes in concrete structural properties at temperatures of 200°F when they prepared this section of the code. The committee did not consider such changes in local properties to be significant to the behavior of the overall structure.

The Southern Nuclear Operating Company response to the NCV is provided in Attachment 1. Attachment 2 contains a discussion of the FNP licensing basis. Attachment 3 provides discussion of questions from Task Interface Agreement 98-11, "Farley's Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures (TAC Nos. MA 4397 and MA 4396)." Attachment 4 provides a description of the RVS structures. Attachment 5 provides a copy of a letter from the code subcommittee chairman providing his review of this issue.

IE01

This letter does not contain NRC commitments. If you have questions please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Dave Morey

EWC/maf:concretempdenialrev6.doc
Attachments:

1. Response to NCV 50-348, 364/00-01-01, Failure to Identify an USQ
2. FNP Reactor Vessel Support (RVS) Temperature Licensing Basis
3. SNC Response to the Evaluation of Task Interface Agreement 98-11
4. Description of the FNP Reactor Vessel Supports
5. Review of NRC Position on RV Support Concrete Temperature (ALA-00-073)

cc: <u>Southern Nuclear Operating Company</u>
Mr. L. M. Stinson, General Manager - Farley

<u>U. S. Nuclear Regulatory Commission, Washington, D. C.</u>
Mr. L. M. Padovan, Licensing Project Manager - Farley

<u>U. S. Nuclear Regulatory Commission, Region II</u>
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

Attachment 1
Response to
NCV 50-348, 364/00-01-01, Failure to Identify an USQ

Attachment 1
Response to NCV 50-348, 364/00-01-01, Failure to Identify an USQ

Violation

NRC Integrated Inspection Report Nos. 50-348/00-01 and 50-364/00-01 section E8 states:

E8.1 (Closed) EEI 50-348, 364/98-05-02: Failure to Identify Defacto 50.59 and Unreviewed Safety Question (USQ) (92903)

a. Inspection Scope (92903)

The inspectors reviewed Escalated Enforcement Item (EEI) 98-05-02 and the NRC staffs response to Task Interface Agreement (TIA) 98-11, "Farley's Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures."

b. Observations and Findings

EEI 98-05-02

As documented in IR 50-348, 364/98-05, in 1977 and again in 1997, the licensee identified that the Reactor Vessel Support (RVS) concrete temperature was greater than the temperature stated in the Updated Final Safety Analysis Report (UFSAR). On December 18, 1997, the licensee approved a change to the UFSAR for the elevated RVS concrete temperatures based on the American Concrete Institute (ACI) codes. The inspectors documented that the licensee did not identify that RVS concrete temperature greater than the temperature stated in the UFSAR was potentially an unreviewed safety question (USQ). As documented in the response to TIA 98-11, the staff concluded that the licensee was not properly applying the ACI codes. The degradation of the RVS concrete as a result of the increased temperature could result in an increase of the RVS malfunction probability as previously evaluated in the UFSAR. 10CFR50.59(c) requires, in part, the holder of a license who desires to make a change in the facility or the procedures described in the safety analysis report which involve an unreviewed safety question, shall submit an application for amendment of his license pursuant to 50.90. 10CFR50.59 (a)(2) states, in part, a proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

Contrary to the above, on December 18, 1997, the licensee approved a change to the UFSAR which failed to identify that allowing higher RVS concrete temperatures was an unreviewed safety question and did not submit an application to amend the license. This NRC identified Severity Level IV violation is being treated as a Non-Cited Violation (NCV) consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is identified as NCV 50-348, 364/00-01-01, "Failure to Identify an USQ" and is in the licensee's corrective action program as Occurrence Report (OR) 1-2000-098.

Admission or Denial

Farley Nuclear Plant is denying the violation.

Reason for Violation

On December 18, 1997, FNP staff approved a change to FSAR section 5.5.14.1.A, based upon the ASME code Section III, Division 2, Subsection CC-3440. The code allowed higher temperature of the concrete supporting the reactor vessel than was indicated by the FSAR. The FNP staff approved the 10CFR50.59 evaluation without identifying the change as an unreviewed safety question (USQ). The NRC staff contends that this change is a USQ as defined by 10CFR50.59 (a) (2) since as stated in the violation, "the increased temperature could result in an increase in the reactor vessel support (RVS) malfunction probability as previously evaluated in the FSAR."

The FSAR description of RVS concrete temperature was identified as being in error as a result of the FSAR verification process being conducted in response to Information Notice 96-17. The temperature of the concrete supports for the RVS in the FSAR was stated as "at or below 130°F." The concrete support temperature was evaluated and determined to be less than 190°F locally and less than 150°F generally. This condition was determined to meet the acceptance criteria of localized concrete temperature of less than 200°F as specified by ASME code Section III, Division 2, Subsection CC-3440, which states in part "the temperature shall not exceed 150°F (66 C) except for local areas, such as around a penetration, which are allowed to have increased temperatures not to exceed 200°F (93 C)." The RVS support temperature was evaluated in the 1970s and determined to have been greater than 150°F locally. The results of this evaluation were not incorporated in the Safety Analysis Reports until the omission was discovered in 1997.

This issue was evaluated by NRC staff as Task Interface Agreement 98-11. As a result of this evaluation the staff concluded that FNP staff had improperly applied the ASME code in that the stated limits should not be applied to principle load bearing concrete. The staff also concluded that "the concrete temperatures in the vicinity of the RVS are and will remain above 190°F" and that "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 to increase." The staff further concluded that "SNC's analysis of the RVS concrete temperature is adequate and that the peak concrete temperature will not exceed 200°F." Based on discussions with the NRC staff it is their belief that FNP does not meet the intent of the ASME code. The staff's argument is based on a presumption that the 200°F limit is not intended to be applied to principal load bearing concrete. The code is silent on application to load bearing structures.

SNC believes that the calculated temperature of the RVS support concrete does meet the requirements of the ACI and ASME code and does not result in an increase in the probability of malfunction of equipment important to safety; therefore, the change to the FSAR does not constitute a USQ. The staff's interpretation that localized temperatures between 150°F and 200°F are not allowed by code is not supported by code statements or industry practice. The ACI 349 and the ASME Code do not specify that the 200°F localized temperature limit should not be applied to load bearing concrete. The Code is silent on the issue of applicability to load bearing or non-load bearing concrete. The NRC Standard Review Plan references ACI 349 as an appropriate code to be applied to PWR primary shield wall, and other safety related concrete structure design. Since the RVS rest on the primary shield wall it is the SNC position that the limit was intended to apply to principal load bearing concrete. In addition, SNC has not found any published NRC exception or clarification to ACI 349 that would indicate the localized temperature limit of 200°F should not be applied to load bearing concrete. Because only a small portion of the primary shield wall is above 150°F, SNC has concluded that this issue is not safety significant. Attachment 2 contains a discussion of the FNP licensing basis for RVS concrete temperature.

SNC has contacted a number of current and former code committee members related to ACI 349 and CC 3440. These members confirm that the 200°F localized limit was not restricted to non-load bearing concrete. The ACI 349 subcommittee chairman on design, at the time paragraph A4.1 was being incorporated into the code, has stated that the code committee was aware that there were small changes in concrete structural properties at temperatures of 200°F. The committee did not consider such changes in local properties to be significant to the behavior of the overall structure.

In summary, this change to the FSAR is within the code allowable limits. Therefore, this change does not involve a USQ and is acceptable as is.

Corrective Steps Taken and Results Achieved

No corrective actions have been taken.

Corrective Steps That Will Be Taken To Avoid Further Violation

No corrective steps are planned.

Date of Full Compliance

FNP has remained in compliance.

Attachment 2
FNP Reactor Vessel Support (RVS) Temperature Licensing Basis

Attachment 2
FNP Reactor Vessel Support (RVS) Temperature Licensing Basis

Both the ASME Boiler and Pressure Vessel Code, Section III, Division 2. Subsection CC-3440 (a) and ACI 349 Appendix A paragraph A.4 state:

> The following temperature limitations are for normal operation or any other long term period. The temperatures shall not exceed 150°F (66 C) except for local areas, such as around a penetration, which are allowed to have increased temperatures not to exceed 200°F (93 C).

The FNP FSAR and the NRC Standard Review Plan (SRP) indicate that the ACI or ASME code applies to the structures internal to the containment. In addition the NRC SER indicated that FNP design of structures internal to containment was in accordance with the Standard Review Plan and was acceptable. The SRP allows the use of the ACI 349 code for the primary shield wall design. Since the RVS rests directly on the primary shield wall it is the SNC position that the code was intended to apply to load bearing structures. The analysis supporting the design application of the FNP reactor vessel supports (RVS) has shown that the localized concrete temperature at the RVS interface will remain below 200°F and complies with the requirements of this code.

Attachment 3
SNC Response to the Evaluation of Task Interface Agreement 98-11

Attachment 3
SNC Response to the Evaluation of Task Interface Agreement 98-11

The following questions and NRR responses were taken from an NRC letter dated January 27, 2000 addressing the questions in Task Interface Agreement (TIA) 98-11. The SNC responses to the first three questions are provided. Question 4 of TIA 98-11 is not discussed since SNC agrees that an adequate temperature analysis was performed.

<u>Response to Questions</u>

SNC's responses to Questions 1 through 3 are as follows:

<u>Question 1</u>.

Is the licensee correct in applying the ACI Code limit of 200°F for localized areas of the reactor vessel supports?

<u>NRR Response to Question 1</u>

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.

<u>SNC Response to Question 1</u>

The initial design of the concrete for the primary shield wall and reactor vessel supports was based on ACI 318-63, Building Code Requirements for Reinforced Concrete. This particular code is silent on the effect of temperature on concrete. The current designs of safety-related concrete structures which support, house, or protect nuclear safety class systems or components or which are component parts of nuclear safety systems are covered by the provisions of ACI 349 "Code Requirements for Nuclear Safety Related Concrete Structures." Hence, ACI 349 is the appropriate code that applies to the primary shield wall and reactor vessel support concrete.

Thermal considerations were first incorporated in the ACI 349 as Appendix A through the 1977 Supplement to ACI 349-76 which was ratified on October 28, 1977. Section A.4.1 of ACI 349 allows concrete temperatures up to 200°F for local areas for normal operation or any other long term period. Per Section A.4.3, evaluation of strength reduction is required only for temperatures higher than those allowed under Sections A.4.1 and A.4.2 of ACI 349. The long term limits are that bulk temperature is not to exceed 150°F and localized temperatures not to exceed 200°F.

The total area of the contact surface of the base of the six supports is less than ten percent of the gross cross-sectional area of the primary shield wall. The calculated temperatures based on air flow distributions at the support/concrete interface for two of the six supports are below 150°F, approximately 165°F at two other supports, and approximately 190°F at the remaining two supports. The temperature in excess of 150°F is limited to four support locations and does not apply across the whole primary shield wall on which the reactor is supported, and is, therefore, a localized effect.

Based upon a very conservative thermal analysis model of the reactor pressure vessel support, which was constructed by Westinghouse, the maximum computed temperature at the interface of the bottom face of the support and the concrete support surface, was determined to be 190°F near the center of the support (Point A, Reference 12) further localizing the peak temperature. The temperature gradient is shown to decrease outwardly from the support center to the extreme edges of the length of the plate (Point B, Reference 12). The analysis assumes no heat transfer into the concrete, but utilizes the bottom of the plate as a boundary condition. Further, the support is welded to the liner plate of the reactor cavity, which would further transfer heat flow into the concrete thus potentially reducing the peak local calculated concrete temperature. Other conservative attributes consist of the actual strength attained in the concrete mass, based on test specimens taken at the time of concrete placement and additional strength gain between the time the concrete was placed and until actual heat up of the nuclear steam supply system (NSSS) occurred.

The USNRC evaluation states that it is 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 bearing for the reactor vessel supports (RVS) loads. However, it is our position that we have properly interpreted and applied the 200°F limit for local conditions because the code provides guidance for conditions of long term temperature application above 200°F. Our position that this condition is local follows the code definition given in the ACI Code (ACI 349-77) Appendix A, "Thermal Considerations." The codes make no distinction based on applied loading or type of structure supported, but only provide an example application of "such as around penetrations." Further, the codes do not restrict the magnitude of loading for local areas with respect to temperatures between 150 and 200°F. However, ACI 349-77, Appendix A "Thermal Considerations," Section A.4.3 provides for justification for structural loading and reduced allowable stresses for long term temperature conditions above 200°F based on concrete testing.

To further substantiate our position, we have discussed the intent of the code for application to principle load bearing concrete with a current member and past Chairman of the ACI 349 subcommittee on design who was the subcommittee chairman in 1976 when paragraph A.4.1 was being incorporated. He agrees with our position and does not support the NRC response that states this local temperature limit should not have been applied to the principle load bearing concrete (Reference 15). Bechtel individuals who also participated in the original development of Section A.4.1 concur with the statements made in Reference 15.

Therefore, it is our position that we have met the intent of the code, since no statement exists within the code that allowable stresses are to be reduced for temperatures between 150 and 200°F.

Question 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?

NRR Response to Question 2

Yes. When SNC determined that the actual concrete temperature 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 10CFR50.59. 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.

SNC Response to Question 2

SNC is properly applying the code limit and RVS temperatures are less than 200°F, therefore no USQ exists. The SNC temperature analysis shows that the temperatures of the supports range from approximately 120°F to 190°F. The analysis did not show that RVS temperatures are and would remain above 190°F.

Question 3.

What are the actual or potential safety consequences for exceeding the ACI code limit of 150°F or 200°F for the RVS concrete.

NRR Response to Question 3

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 5 x 10 EE 19 n/cm and 10 EE 19 rads, respectively.

SNC Response to Question 3.

Response to Question 3 is provided in two parts as follows:

a) Temperatures of greater than 200°F: Because the thermal analysis is of a very conservative nature which tends to over predict the localized long term temperatures at the concrete/base plate bottom interface, our position is that this condition does not apply to Farley Nuclear Units 1 and 2. Therefore, our response is limited to the actual analyzed conditions of service for Units 1 and 2, which pertains to a temperature range of 200 degrees Fahrenheit and lower.

b) Temperatures between 150 and 200°F: Although it is realized that some mechanical properties of the concrete mass may change based on long term exposure to this temperature range, we believe that the overall structural strength of the structure (primary shield wall) will not be affected to the extent that there is any loss of intended function for this temperature range. Our position is based on (1) the fact that the shield wall is a massive structure of which only a small percentage of the gross area is loaded by the Reactor Vessel Support, (2) test specimens of concrete tested at the time of concrete placement significantly exceed the specified strength, (3) strength gain subsequent to placement and testing continued on a predictable basis from the time of placement until the time of original heat application to the concrete (4) the actual volume of concrete reaching higher temperatures is further localized on the surface area of the plate, and (5) the bearing surface of the concrete is essentially confined which results in high allowable bearing stress.

We also observe that mechanical properties of concrete due to long term exposure to temperatures between 150 and 200°F initially decrease but tend to stabilize after a period of time (Reference 13).

Therefore, based on consideration of potential changes in concrete mechanical properties, these changes would be adequately compensated by the factors cited above. It is our understanding that the knowledgeable individuals who prepared the code requirements also knew and considered the potential changes in local concrete properties and did not intend to limit the applicability of section A.4.1 of the code to non-principal load bearing structures (Reference 15). Nor was it the intent of the code to imply that a reduction in allowable stress was advisable or required for local areas of principal load bearing structures with a temperature range up to 200°F.

SNC CONCLUSION

We conclude that the peak calculated local concrete temperature of 190°F at the support base/concrete interface is acceptable. The requirements of Subsection CC-3440 of ASME Section III, Division 2, are in agreement with the concrete temperature limits specified in Section A.4 of ACI-349-76 (1977 Supplement), and are therefore acceptable.

REFERENCES

1. Farley Nuclear Plant FSAR.
2. Calculation 9.1-11, Reactor Vessel Support Concrete Temperatures.
3. ASME Section III, Division 2 (ACI 359-74), Code for Concrete Reactor Vessels and Containments, 1975.
4. ACI 349-76 (with 1977), Supplement), Code Requirements for Nuclear Safety Related Concrete Structures.
5. Drawing U-166490, Revision E, Reactor Vessel Supports.
6. Drawing D-206204, Revision 3, Plan and Sections, Primary Shield Walls
7. Drawing D-206271, Revision 5, Plan and Sections, Sheet 1, Containment.
8. Drawing D-176204, Revision 9, Plan and Sections, Primary Shield Walls.
9. Drawing D-176271, Revision 6, Plan and Sections, Sheet 1, Containment.
10. Drawing U-207668, Revision B, Reactor Vessel Supports.
11. ACI 318-63, Building Code Requirements for Reinforced Concrete.
12. Sketches Primary Shield Wall and Reactor Vessel Supports, Plan and Sections, 3 sheets.
13. ORNL/NRC/LTR-94/22, "Summary of Materials Contained in the Structural Materials Information Center", Oak Ridge National Laboratory, November 1994.
14. W-1332, June 24, 1977.
15. ALA-00-073, Review of NRC Position on RV Support Concrete Temperatures, May 25, 2000

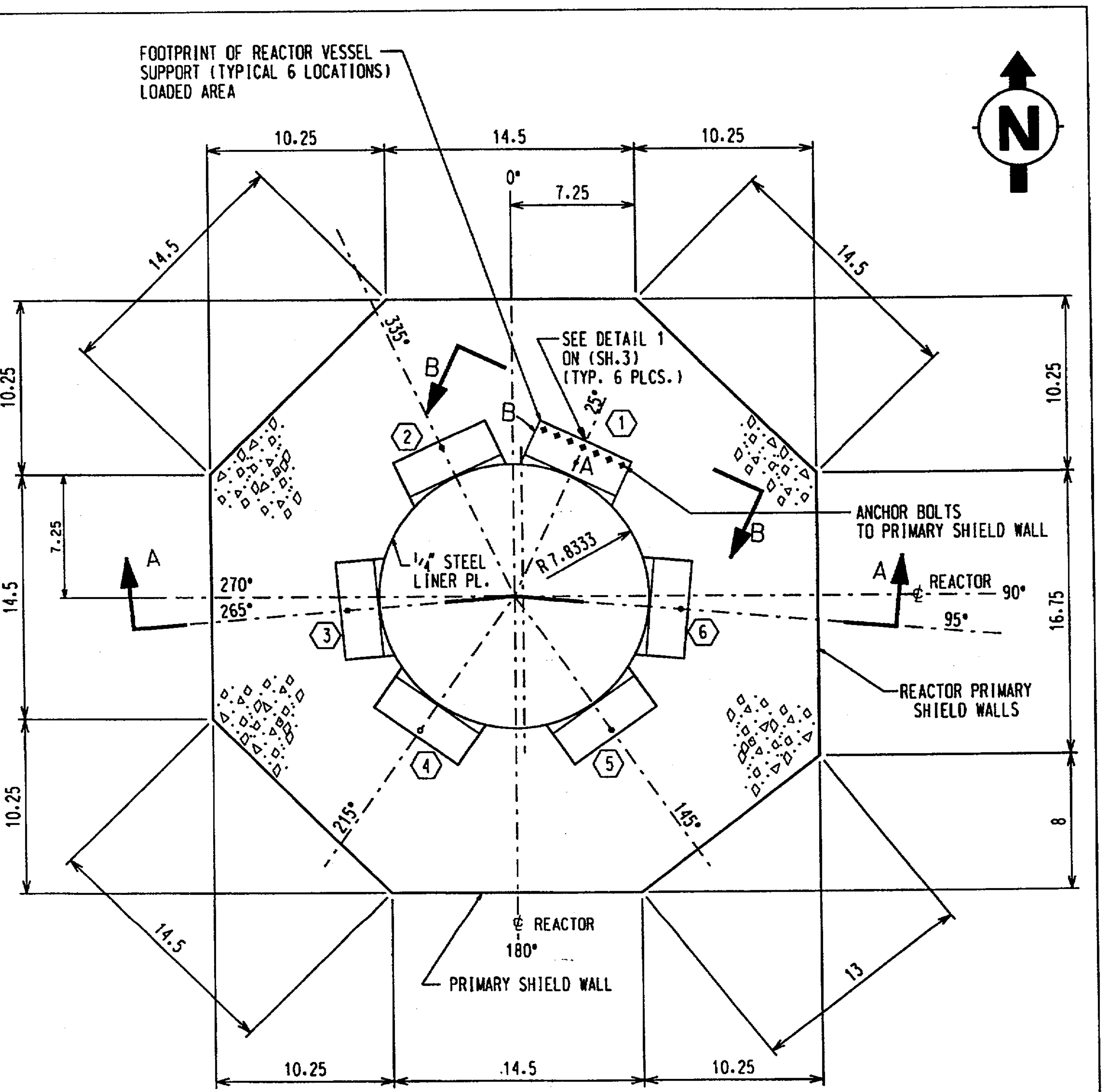
Attachment 4
Description of the FNP Reactor Vessel Supports

Attachment 4
Description of the FNP Reactor Vessel Supports

DESIGN DESCRIPTION

The primary shield wall provides shielding for the Reactor Pressure Vessel (RPV). The primary shield wall also provides support for the RPV. The primary shield wall is a massive heavily steel reinforced concrete structure having walls approximately 9.7 feet in thickness. It is octagonal in shape on the exterior perimeter with a circular shaped center cavity. The height of the primary shield wall is 12.4 feet, from the elevation at which the RPV rests to the top of the containment base slab, and continues an additional 34.25 feet from the top of the containment base slab to the bottom of the reactor cavity base slab. The center cavity is lined with steel plate material which is also welded to the bottom part of the RPV supports.

The Reactor Pressure Vessel (RPV) rests on six steel supports which are located underneath the RPV nozzles. There are six supports; one support for each of the three hot leg nozzles and one support for each of the three cold leg nozzles. Each nozzle support consists of two parts, one part (the upper part) which is attached to the nozzle and the lower part which supports the upper part and is in turn supported by the concrete primary shield wall. The lower part is anchored into the concrete primary shield wall. The upper part is allowed to slide on the lower part to allow for thermal expansion of the RPV. The load path from the RPV is the nozzle support upper part through the lower part to the concrete surface. The gross cross-sectional area of the primary shield wall is approximately 835 square feet, as compared to sum of the footprint areas of the six supports, which is approximately 76.6 square feet. Based on these areas, the supports occupy less than ten percent of the cross-sectional area of the primary wall.



ALL DIMENSIONS ARE IN FEET.

PLAN @ REACTOR VESSEL SUPPORTS

UNIT 1 AS DRAWN

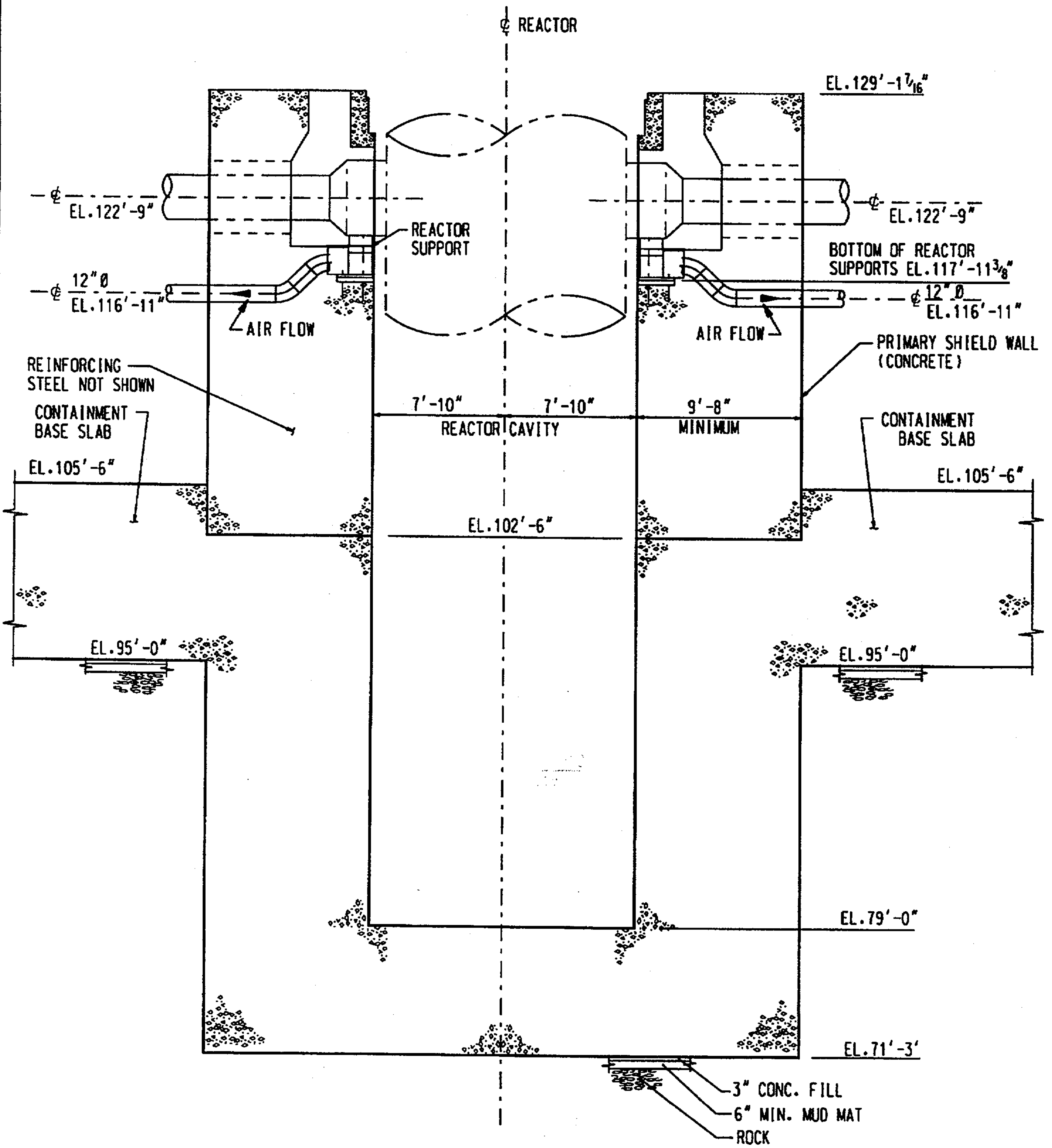
UNIT 2 OPP HAND

SUPPORT NO.

REACTORCAVITY.DGN

PRIMARY SHIELD WALL & RPV SUPPORTS

REFERENCE 12, SHEET 1 OF 4

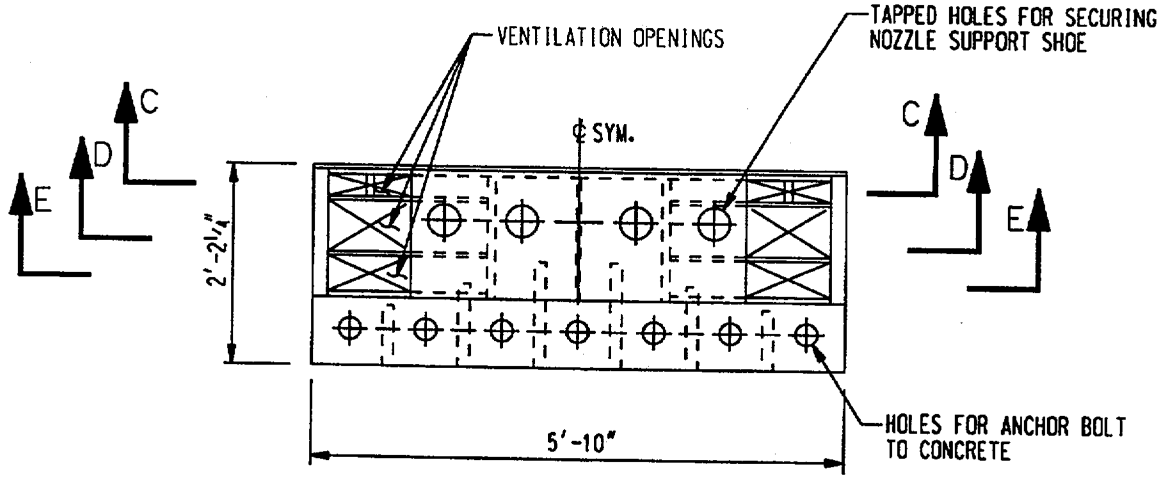


SECTION A-A

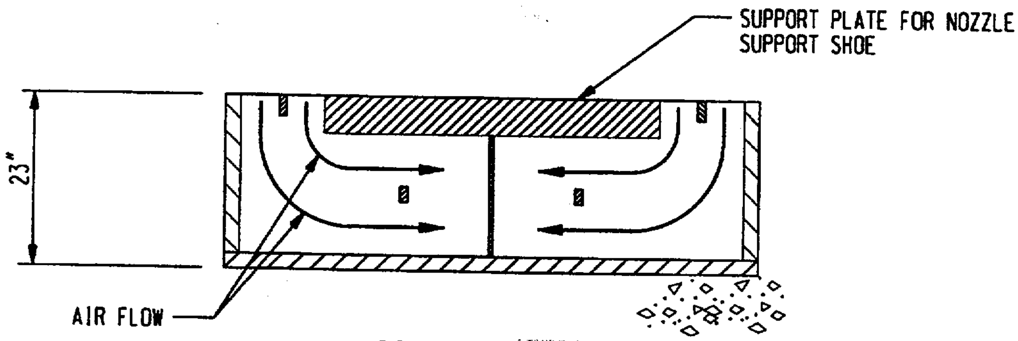
REACTORCAVITY.DGN

PRIMARY SHIELD WALL & RPV SUPPORTS

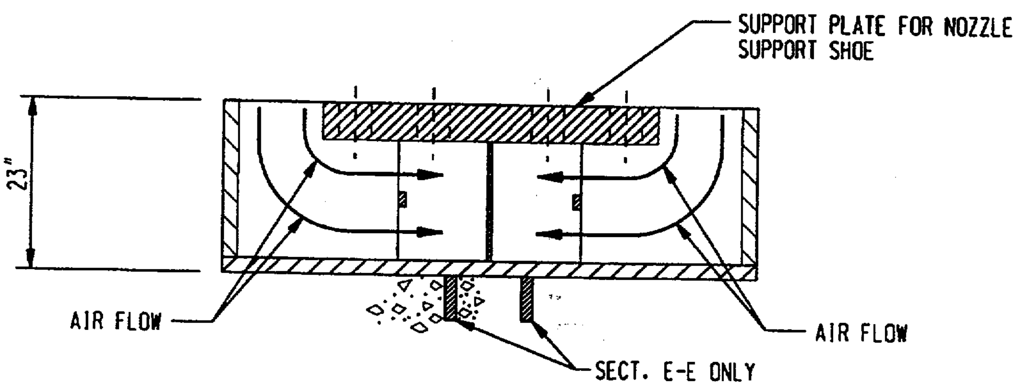
REFERENCE 12, SHEET 2 OF 4



DETAIL 1



SECTION C-C

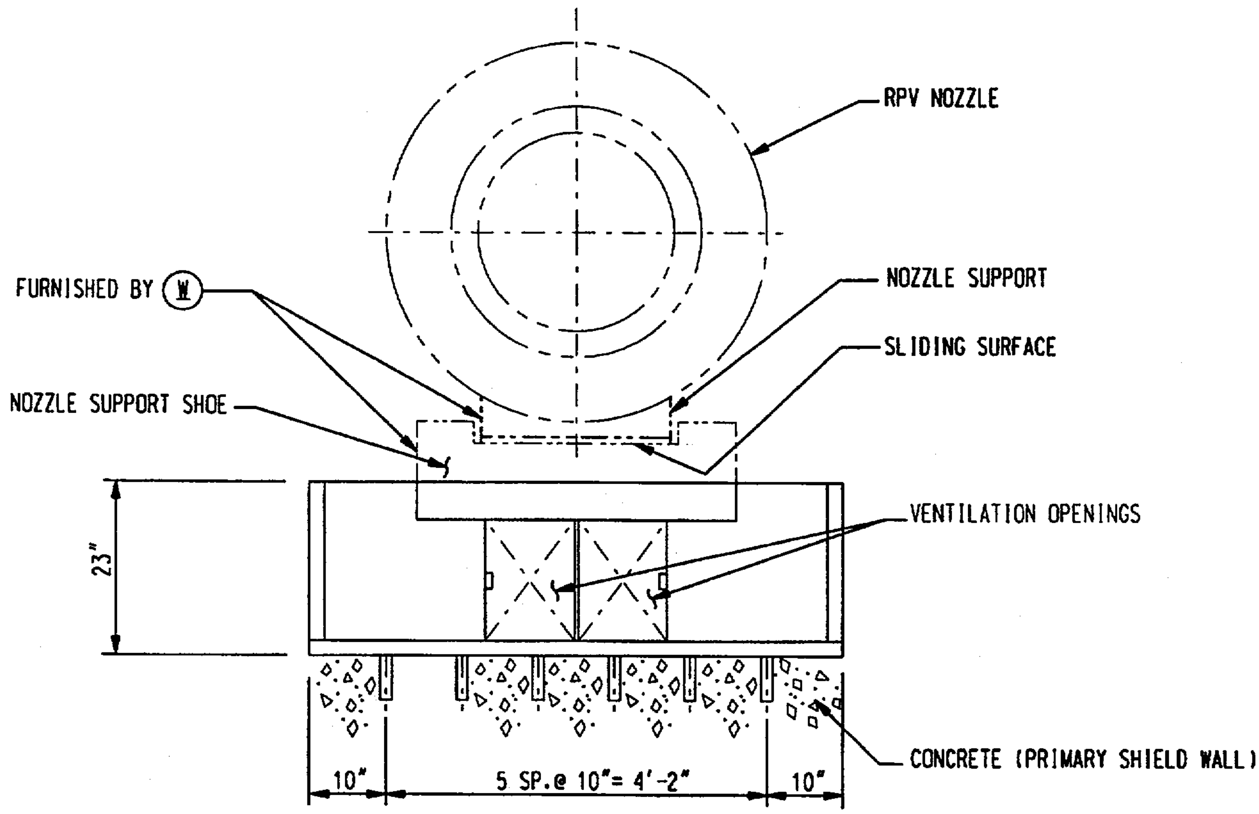


SECTION D-D (AS SHOWN)
SECTION E-E (SIMILAR)

REACTORCAVITY.DGN

PRIMARY SHIELD WALL & RPV SUPPORTS

REFERENCE 12, SHEET 3 OF 4



SECTION B-B

(NOZZLE SUPPORT, NOZZLE SUPPORT SHOE
AND STRUCTURAL STEEL SUPPORT TO CONCRETE SURFACE)

ALL PIECES ARE STEEL PLATE OR STEEL BAR

APPROXIMATE FLOW PER SUPPORT BASED ON ORIGINAL STARTUP TESTS		
SUPPORT NO.	FLOW (CFM)	PEAK CALCULATED TEMPERATURE °F
1	≈ 2150	≈ 183
2	≈ 2600	≈ 165
3	≈ 4700	≈ 120
4	≈ 3250	≈ 140
5	≈ 2700	≈ 160
6	≈ 2000	≈ 190

REACTORCAVITY.DGN

PRIMARY SHIELD WALL & RPV SUPPORTS

REFERENCE 12, SHEET 4 OF 4

Attachment 5
Review of NRC Position on RV Support Concrete Temperature
(ALA-00-073)

ALA-00-073

Westinghouse
Electric Company, LLC

Box 355
Pittsburgh, Pennsylvania 15230-0355

May 25, 2000

Mr. D. N. Morey, Vice President
Farley Project
Southern Nuclear Operating Company
P. O. Box 1295
Birmingham, Alabama 35201-1295

Attention: Mr. Mark J. Ajluni

SOUTHERN NUCLEAR OPERATING COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT
UNITS 1 AND 2
Review of NRC Position on RV Support Concrete Temperatures

Dear Mr. Morey:

Attached for your information and use is a summary report prepared by Mr. Richard S. Orr of Westinghouse relative to his review of the NRC document entitled "Evaluation of Task Interface Agreement 98-11, Farley Nuclear Plant Units 1 and 2, Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures".

Should you have any questions, please feel free to contact Mr. Orr (412) 374-5924 or me.

Very truly yours,

WESTINGHOUSE ELECTRIC COMPANY

T. W. Wallace, Manager
Southern Nuclear - Farley Projects

/attachment

Attachment to ALA-00-073

<u>Review of NRC Letter " Evaluation of Task Interface Agreement 98-11"</u>

Paragraph A.4.1 of ACI 349 states:

> "**A.4.1** The following temperature limitations are for normal operation or any other long term period. The temperatures shall not exceed 150 F except for local areas, such as around penetrations, which are allowed to have increased temperatures not to exceed 200 F."

I was Chairman of the ACI 349 subcommittee on design when we included this paragraph in 1976. The paragraph is identical to that included in Division 2 of ASME Section III for Containment Vessels.

The discussion of the effect of temperature on structural properties contained in the NRC evaluation is reasonable (10% reduction in compressive strength, 30% reduction in modulus of elasticity, 22% reduction in Poisson's ratio). The reductions would be less if the concrete is subjected to significant dead load as stated in NRC's evaluation. Tests by Abrams, reported in ACI Committee report ACI 216R-89, show zero reduction in compressive strength for specimens loaded to 40% of the concrete strength.

The Code committee was aware that there were small changes in structural properties at temperatures of 200 degrees at the time that they included paragraph A4.1 in the Code. The committee did not consider such changes in local properties to be significant to the behavior of the overall structure. The paragraph, as written, permits local temperatures immediately below the supports up to 200 degrees F. I do not agree with the NRC response R1 that states that this local temperature limit should not have been applied to the principal load bearing concrete.

The concrete immediately below the support will have a significant increase in concrete bearing strength due to the confinement provided by the steel support. Only partial credit is given to this increase in the bearing strength permitted by ACI 349. The limiting section would normally be at some distance below the steel support where the temperature of the concrete would also be lower.

Richard Orr

Richard S. Orr
Westinghouse Electric Company