PDR-50-482

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 27 1979

Ms. Maude Skillman
Corresponding Secretary
Topeka Branch, Women's
International League for
Peace and Freedom
1116 West Fifth
Topeka, Kansas 66606

Dear Ms. Skillman:

This is in response to your letter of December 24, 1978, to the Nuclear Regulatory Commission expressing your concern for the quality of cement being used at the Wolf Creek Nuclear Plant. Your letter to President Carter on the same date and expressing the same concern was also referred to this office for response.

Your letters indicated that while you were aware of the NRC's investigations into reported deficiencies, including the questions on cement quality being used at the Wolf Creek Nuclear Plant, we should be cognizant of the fact that faulty cement was the cause of the April 1978, Willow Island, West Virginia cooling tower construction accident at a fossil fueled power plant. The NRC staff has been aware of the information released by the Occupational Safety and Health Administration (OSHA) regarding the Willow Island construction accident. As you probably know, there were in general three main reasons noted by OSHA as contributors to that failure:

1. Proper tests were not conducted on the concrete prior to removal and moving of forms.

2. The scaffold formwork system was not properly secured to the completed tower sections.

3. The concrete transport system used to lift plastic concrete to the top of the tower was not anchored and maintained to support the maximum intended load.

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We are not aware that faulty cement was a contributor to the accident at Willow Island. In discussing this issue we refer to cement as the dry material which when reacting with water will harden and bond other materials such as aggregates into a matrix known as concrete. Charges by one of the contractors involved in the construction at Willow Island have alleged "defective concrete," apparently meaning the concrete did not gain strength as rapidly as had been expected. This relates to Item 1 above in the OSHA findings.

In the case of the Wolf Creek Nuclear Plant our investigations have specifically highlighted among other things, a search for any facts which would indicate faulty cement. Each avenue we have explored on this subject to date has indicated that the cement met all necessary requirements, but certain cement samples tested by cube strengths did indicate lowered strength characteristics. This coupled with a mix design (the proportioning of each of the ingredients for concrete - cement, water, sand and gravel) which was not performed in the manner specified by the architect-engineer appears to have been a contributing reason for the apparent low strength observed in some test concrete cylinders. More data is being developed and will be evaluated to determine whether there are any identifiable cement deficiencies.

We are enclosing for your information a current summary of significant facts related to the concrete problems at the Wolf Creek site. At the present time no concrete is being placed in the reactor containment building pending resolution of the base mat strength, except for repair work.

Copies of the investigation report are available in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the local public document room for the Wolf Creek plant located at Burlington, Kansas.

Sincerely,

John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosure:
Summary of Concrete Problems
on the Wolf Creek Plant

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Summary of Concrete Problems
Wolf Creek Nuclear Plant
March 1979

Concrete was placed for the reactor containment building base mat in a continuous operation on December 12 and 13, 1977. The total volume of the 10 foot thick mat was nearly 6600 cubic yards. Sample test cylinders of the concrete were taken during the placement and subsequently tested at 7 and 28 days after placement to determine the rate of strength gain. Sample cylinders for the final 90-day strength determination also were obtained. On March 13, 1978, the 90-day cylinders were tested---about 9% failed to meet one test criterion; about 50% failed to meet a second test criterion. The NRC inspector was informed of the apparent low cylinder strength on March 15, 1978. Inspection Report STN 50-482/78-04, dated March 31, 1978, noted that the question of the 90-day strength of the concrete for the reactor containment base mat had not been settled.

The licensee, Kansas Gas & Electric Company, informed the NRC on May 3, 1978 that, in the licensee's opinion, the apparent low test results of the concrete strength were not required to be reported to the NRC. The licensee agreed to send the NRC for its information, a report about the licensee's investigations which had been initiated. The NRC was provided interim reports, dated May 3, 1978, May 25, 1978 August 18, 1978, and September 29, 1978, about the progress of the licensee's investigation. The licensee's final report was submitted on October 26, 1978.

Review of the final report by the NRC raised questions about the conclusions contained in the report. On November 13, 1978, the Region IV (Dallas) Office of the NRC, with the assistance of a consultant, began an investigation into the apparent low strength concrete test cylinders. By December 1, 1978, the investigators had concluded from the information available that the specifications the licensee had established for acceptance of the concrete had not been met and that the reactor containment base mat strength was in question. The NRC preliminary evaluation of the base mat concrete strength based on the test cylinders indicated a value about 10% below the intended strength of 5,000 pounds per square inch (psi).

On December 5, 1978, a meeting was held by the Director of NRC Region IV with the licensee to discuss the status of the investigation and to emphasize the importance placed on this problem by the NRC.

On December 13, 1978, the licensee reported another concrete deficiency, a through-wall void in the concrete wall beneath the equipment hatch in the reactor containment building. Another void was found beneath the personnel lock, but was not a through-wall void. In a letter issued on

December 19, 1978, the NRC, through its Region IV Office, informed Kansas Gas & Electric Company of the NRC's concerns regarding the concrete problems and the actions that the licensee was to address in order to satisfy these concerns. The concerns related to the overall quality assurance program including controls and procedures related to concrete placement, quality control, inspection, testing and qualification of personnel, as well as the independence of the inspection and verification organizations. The NRC also confirmed a commitment by the licensee to stop the placement of concrete in safety related structures until the quality assurance matters outlined in the letter were corrected and demonstrated to the satisfaction of the NRC.

On January 4, 1979, a meeting was called by the NRC to discuss the findings of the NRC investigation and the position of the licensee on those findings. The meeting, held in Bethesda, Maryland, included representatives of all involved parties and members of the public and the news media.

As a result of the meeting, the licensee initiated additional testing on cube samples stated to have been cut from the remains of the original 90-day test cylinders. The licensee submitted a report on February 28, 1979, describing the results of these additional tests. That report is currently being evaluated. The NRC in a letter dated February 8, 1979, requested that the licensee consider cut cube sample testing on remnants of 28-day test cylinders and that an assessment of the concrete strength be made using the test data obtained from all of the test cylinders. It was also requested that the value for the strength obtained be used to evaluate the load carrying capacity of the structure for the required loading combinations. The licensee's response to these items has not yet been received.

Region IV, after additional inspections at the site during February 1979, concluded that the licensee had satisfactorily met the commitments agreed to in the December 19, 1978 letter. On March 5, 1979, another letter was issued by Region IV which called for no further placement of concrete in the reactor containment building until the question on the acceptability of the base mat has been resolved. The licensee will, however, complete the necessary repairs to the voids in the reactor containment wall.

The licensee resumed placement of safety related concrete except for the reactor containment on March 6, 1979. On March 8, 1979, the licensee stopped work on safety related concrete after licensee quality control personnel observed that concrete was being moved by vibrators over a greater horizontal distance than permitted by the governing code. This deficiency was observed during the placement of a wall section of the auxiliary building. The initiative for the stop-work action was taken by the licensee. The licensee lifted the stop-work order on March 22, 1979, relative to placement of safety-related concrete except for difficult placements and concrete in the reactor containment building.

Until results are received from the licensee relating to a structural evaluation using the actual test strength of the 90-day test cylinders, no final determination can be made on the acceptability of the base mat. It should be noted that the need for 5000 psi strength concrete was determined by the licensee's architect-engineer and the value is not an NRC requirement. Typical base mat concrete strengths at other nuclear facilities have been specified at 3000 and 4000 psi at 28-days while others might require 6000 psi at 90-days. Needed concrete strength at a specific site must be consistent with the soil conditions and the specific structural loadings at each individual site.

There has been considerable public interest in this case resulting in several requests to the NRC for suspension or revocation of the permit to construct this plant. These requests still remain for final action.

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