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March 28, 1994
Fort St. Vrain
Unit No. 1
P-94022

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Docket No. 50-267

SUBJECT: **QUARTERLY SUBMITTAL OF THE 10 CFR 50.59 REPORT OF CHANGES, TESTS AND EXPERIMENTS FOR FORT ST. VRAIN DECOMMISSIONING**

REFERENCE: NRC Letter dated November 23, 1992, Erickson to Crawford (G-92244)

Gentlemen:

This letter transmits the quarterly 10 CFR 50.59 Report of Changes, Tests, and Experiments affecting Decommissioning of the Fort St. Vrain (FSV) Nuclear Station. The attached report includes a description of each change, test and experiment as well as a summary of the safety evaluation. This report covers the period of November 16, 1993 through February 15, 1994.

This report is being submitted pursuant to Condition (b)(2) of the "Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility", transmitted in the referenced letter, which states the following:

> "The licensee shall submit, as specified in 10 CFR 50.4, a report containing a brief description of any changes, tests and experiments, including a summary of the safety evaluation of each. The report must be submitted quarterly."

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If you have any questions concerning this report, please contact Mr. M. H. Holmes at (303) 620-1701.

Sincerely,

D. W. Warembourg

D. W. Warembourg
Decommissioning Program Director

DWW/JRJ

Attachment

cc: Mr. John H. Austin, Chief
Decommissioning and Regulatory
Issues Branch

Regional Administrator, Region IV

Mr. Robert M. Quillin, Director
Radiation Control Division
Colorado Department of Health

MARCH 1994
QUARTERLY 10 CFR 50.59 REPORT OF CHANGES, TESTS AND EXPERIMENTS
FOR FSV DECOMMISSIONING

Background:

The following is a brief discussion of 10 CFR 50.59 changes to the Fort St. Vrain (FSV) facility or procedures as described in the Decommissioning Plan (DP) and tests and experiments not described in the DP, in the time period from November 16, 1993 through February 15, 1994.

While this report is similar to past reports of changes, tests and experiments submitted in accordance with 10 CFR 50.59, the quarterly decommissioning reports are submitted pursuant to Paragraph (b)(2) of the FSV Decommissioning Order (issued in NRC letter dated November 23, 1992, Erickson to Crawford (G-92244)), which states:

> "The licensee shall submit, as specified in 10 CFR 50.4, a report containing a brief description of any changes, tests and experiments, including a summary of the safety evaluation of each. The report must be submitted quarterly."

<u>Changes to the FSV Facility or its Procedures as Described in the Decommissioning Plan</u>

There were no procedure changes during this reporting period which represented changes to procedures as they are described in the DP. Descriptions of changes to the facility as described in the DP are as follows:

1. Removal of Core Support Floor Upper Insulation in Parallel With Activated Graphite Removal

DP Section 2.3.3.10 discusses removal of the insulation on top of the Core Support Floor (CSF), shown in DP Figure 3.1-30. The DP indicates that prior to removal of the insulation on top of the CSF, it was planned to first complete the removal of all core components and the core barrel to within a few feet of the CSF, with the PCRV shield water level then lowered to just above the top surface of the CSF. It was also planned that an electro-hydraulic ram hoe would be lowered into the PCRV to break up the silica blocks, which provide most of the insulation on top of the CSF.

This change involve an alternate sequence and method for removing the insulation on t of the CSF. This alternate method does not exclude possible use the ram hoe method, described in DP Section 2.3.3.10.1. The alternate sequence enables CSF upper insulation to be removed in conjunction with removal of the core components and the core barrel, while PCRV shield water level remains high above the CSF to provide shielding for the activated graphite blocks and core barrel. The alternate method involves the use of long handled tools and/or remote actuated tooling to remove the CSF upper insulation, without breaking up the silica blocks.

The safety evaluation determined that the proposed change in the sequence and methodology for removal of the CSF upper insulation does not involve an unreviewed safety question. The probability of an accident or malfunction of equipment important to safety previously evaluated in the DP is not increased since packaging the insulation in whole blocks, as opposed to pieces broken by the ram hoe, would not increase the number of handling evolutions or crane operations involving the insulation. The consequences of postulated drop of a package containing CSF upper insulation are not impacted by the revised sequence and methods, and the consequences were shown to be bounded by the postulated drop of activated graphite blocks, previously evaluated in the DP. No new failure modes are created by the revised methods, and no new accidents or malfunctions are created. The safety evaluation considered the possibility of insulation floating to the top of the shield water, and determined that radiation protection personnel would be able to identify any necessary precautions so workers could safely remove the insulation, with no new accidents involved. The operations involved in this alternate sequence and methodology do not impact the basis of any Decommissioning Technical Specification.

2. Transfer of a Basket of Activated Graphite Blocks from the Reactor Building Refuel Floor to the Hot Service Facility Without the Shield Bell

DP Section 2.3.3.8.5, "General Graphite Block Removal Sequence", discusses the general sequence of operations used in removal of graphite blocks. Under normal conditions, baskets containing activated graphite blocks are removed from the PCRV using the shield bell, and transferred from the shield bell to a cask liner in the Hot Service Facility (HSF), where the shipping package is made ready for shipment. DP Section 2.3.3.8.5(6) also considers possible transfer of a cask liner containing activated graphite blocks from a temporary storage cask to the transport cask, stating: "Any necessary shielding will be provided during the transfer of the liner from the storage cask to the shipping cask."

This change involved unshielded transfer of a basket of activated graphite blocks from the Reactor Building refuel floor to the HSF. Activated graphite blocks were loaded into shipping cask basket No. 63 in the PCRV, transferred in the shield bell to the Reactor Building refuel floor for a drying period, then moved in the shield bell to the HSF where the basket was to be lowered into a shipping cask liner. As basket No. 63 was being lowered into the HSF, it was detected that the basket tilted, indicating that one of the three pickup points on the basket was not fully engaged with the shield bell/basket lifting mechanism. The basket was therefore returned in the shield bell to the Reactor Building refuel floor, and the proposed transfer of the basket to the HSF using a basket grappling mechanism without the shield bell was evaluated.

The safety evaluation determined that the one time operation of unshielded transfer of a basket of activated graphite blocks to the HSF did not involve an unreviewed safety question. The primary concerns were ALARA/Radiation Protection and the potential for accidentally dropping/tipping the basket. The FSV ALARA Committee and the Work Package Change Notice addressed the ALARA concerns. The rigging was specified such as to minimize the possibility of dropping or tipping the basket, and was equal to or better than the normal activated graphite block basket handling rigging. The Work Package Change Notice included requirements to inspect the lifting rods of the transfer basket for damage. Since the basket lifting equipment and handling rigging were rated for the weight and configuration of the basket, the probability of an accident or malfunction previously evaluated in the DP, such as drop of activated graphite blocks, was not increased. The curie content of the graphite contained in basket No. 63 was well below that assumed in Section 3.4.5 of the DP for postulated drop of activated graphite, and potential drop heights of this basket were much lower than assumed in the DP. Therefore, the consequences of a drop accident or malfunction would not be increased above those previously evaluated in the DP. This transfer evolution did not raise the potential for accidents different from those previously evaluated in the DP. No Technical Specification margins of safety were reduced as a result of this transfer evolution.

Tests or Experiments not Described in the Decommissioning Plan

No tests or experimerts have been conducted during this reporting period that are not described in the DP.