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September 15, 1994
Fort St. Vrain
Unit No. 1
P-94077

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 May 16, 1994 through August 17, 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

SEPTEMBER 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 May 16, 1994 through August 17, 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), 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."

Changes to the FSV Facility or its Procedures as Described in the Decommissioning Plan

Brief descriptions of changes to the facility and procedures as described in the DP and a summary of the associated safety evaluations follow:

1. PCRV Shield Water Releases

The original plan for discharging PCRV shield water is described in DP Section 3.3.2 and Section 4.2 of the Environmental Report Supplement (ERS) for FSV Decommissioning (Reference 1). The original plan was to initially transfer the PCRV shield water to a liquid waste holdup tank in the existing Radioactive Liquid Waste System (System 62) for sampling and analysis. Water would then be discharged from System 62 in accordance with the requirements of the FSV Offsite Dose Calculation Manual (ODCM). Since the capacity of a System 62 transfer pump is 10 gpm, and minimum dilution flow is 1100 gpm, the DP and ERS indicate that a minimum dilution factor of 110 would be achieved. After the water in the PCRV had been processed to the extent that the concentrations of Co-60, Cs-137 and Fe-55 in the entire PCRV water volume were less than approximately 1% of the 10 CFR 20 MPC limits, and tritium

concentration less than MPC, it was planned to send the water from the PCRV directly to the discharge line.

The current plan for discharging PCRV shield water involves use of the Reactor Building Sump (RBS), which has a much greater volume than the System 62 tanks, permitting more rapid decreases in the PCRV shield water level. Higher than expected radiation levels from various components in the PCRV have dictated maintaining a high PCRV water level throughout decommissioning operations to date, to ensure that occupational radiation exposures were maintained ALARA. As a result, the PCRV shield water level has not been gradually lowered as originally planned and it is now desirable to rapidly lower the PCRV shield water level.

Shield water will be transferred from the PCRV, through the PCRV shield water demineralizers, into the rectangular portion of the RBS (in which a liner is installed to prevent possible contamination of the RBS concrete), and to the RBS pumps' suction. It is anticipated that concentrations of some radionuclides in the shield water transferred to the RBS will be greater than 1% of their 10 CFR 20 MPCs. A dilution factor of 110 may not be achieved during release since each of the two RBS pumps has a capacity of approximately 60 gpm. The higher radionuclide concentrations, lower dilution factor and use of the RBS instead of a System 62 holding tank represent deviations from the DP description of PCRV shield water release operations. Therefore, a safety evaluation was required and a pathways analysis was performed to conservatively estimate doses to the public that could result from using the RBS to discharge shield water.

In order to perform the pathways analysis, a test was conducted to determine the curie concentration of the shield water after it had passed through the shield water system demineralizers, simulating the expected concentrations to be released to the RBS. Using the calculation methodology from the FSV ODCM and the conservative assumption of 500,000 gallons of water (approximately 310,000 gallons are currently in the PCRV) released over 167 hours, resultant offsite doses were calculated to be about 30% of those presented in Section 4.2.6.7 of the ERS (for the 500 curies tritium case). Although radionuclide concentrations higher than the 1% of MPC assumed in the previous pathways analysis were used for some gamma emitting radionuclides, lower doses were calculated since tritium concentrations in the PCRV are much lower than previously assumed. The radionuclide concentrations used in these pathways calculations will be proceduralized prior to releases to assure offsite doses do not exceed those calculated. All releases from the RBS will be controlled in accordance with the FSV ODCM. Offsite doses to the general public will be in compliance with 10 CFR 20, 10 CFR 50 Appendix I, and the EPA Safe Drinking Water Standards in 40 CFR 141.

The worst case accident that could occur during transfer of shield water into the RBS would involve spilling the entire contents of the PCRV into the Reactor Building. A bounding accident of this type is evaluated in DP Section 3.4.7, "Loss of PCRV Shielding Water Accident", which conservatively postulates 423,500 gallons of shield water containing 100,000 curies of tritium (62.4 μCi/cc) is dumped into the Reactor Building basement, with tritiated water evaporating to the atmosphere. Since the PCRV shield water actually contains less than 5 curies of tritium, with tritium concentrations at approximately the MPC of 3 E-3 μCi/cc, it is not considered possible for an accident involving transfer of PCRV shield water to the RBS to exceed the consequences of the accident previously evaluated in DP Section 3.4.7.

The probability of an accident or malfunction previously evaluated in the DP is not being increased since the volume of water to be transferred to the RBS at any one time will be much less than 423,500 gallons, with tritium concentrations well below 62.4 μci/cc and no increase in the number of PCRV discharge operations. The automatic monitoring and protection features which governed releases from the RBS and System 62 during operations will also govern the releases of shield water from the RBS, since the release path and associated protective equipment have not changed. No equipment classified important-to-safety will be affected by this activity.

Discharging shield water from the RBS does not create the possibility of new types of accidents or malfunctions not previously evaluated. The evolution does not create the potential for an uncontrolled release of radioactive liquid effluent, since the normal release path will be used for these discharges in accordance with the FSV ODCM. No new failure modes are introduced.

No margins of safety defined in the bases of Technical Specifications are reduced. The evaluation for the effects of discharging shield water from the RBS on the DP accident analyses and ERS has taken into account the applicable Technical Specifications and has bounded the conditions under which the specifications permit safe decommissioning operations. A program is in place at FSV conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program is contained in the FSV ODCM, is implemented by procedures, and includes remedial actions to be taken whenever the program limits are exceeded.

Based on the above, the safety evaluation concluded that the planned method for discharging PCRV shield water from the RBS does not constitute an unreviewed safety question.

2. Modification to the Radioactive Liquid Waste System to Provide Recirculation and Purification Capability for Water in the Reactor Building Sump

Engineering Change Request ECR-94-004 authorized a modification to the line that connects the discharge of the Reactor Building Sump (RBS) pumps to the Radioactive Liquid Waste Sump. A tee was installed in the line to enable water from the RBS to be pumped through the PCRV shield water demineralizers and returned to the RBS. This path can be used to further reduce the concentration of radionuclides in the RBS water in preparation for discharge of the contents of the RBS. An isolation valve was installed in the branch line from the tee that is connected to the shield water demineralizers' inlet header by 1-1/2" diameter high pressure hose.

The safety evaluation determined that the probability or consequences of accidents or malfunctions previously evaluated in the DP are not increased by this modification. Additional tubing is equivalent to that currently used in the PCRV shield water system, so the probability of a loss of shield water accident (analyzed in DP Section 3.4.7) is not affected. The small amount of additional piping added by this activity can be isolated using the new valve. The additional recirculation line added by this modification will be procedurally controlled to prohibit the accidental discharge of radioactive liquids. The recirculation water will be routed back to the RBS. Any potential mishaps associated with this RBS recirculation line are enveloped by the accident analysis in DP Section 3.4.7. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the DP has not been created. The Decommissioning Technical Specifications refer to the ODCM for radioactive effluent discharges. All radioactive effluents, including those from the RBS, are discharged in accordance with the requirements of the ODCM, which assures compliance with the applicable regulatory requirements. The margins established in the ODCM to assure regulatory compliance are not affected by this modification.

Based on the above, the safety evaluation concluded that the modification does not constitute an unreviewed safety question.

3. Removal of Steam Generator Primary Assemblies

DP Section 2.3.3.11.1 states that the steam generator (SG) primary assemblies will be removed from the PCRV, washed down, wrapped in poly film or Herculite, and transferred to the reactor building truck bay for packaging in a culvert that has been cut in half axially. Grout could be pumped into a SG primary assembly and/or inside the culvert around the outside of an assembly's jacket to provide shielding as needed. An alternative was evaluated, in accordance with current planning, in which the SG primary

assemblies are lifted from the bottom of the PCRV, washed down, and packaged on the ledge inside the PCRV (where the Decommissioning Rotary Work Platform currently rests) or on the refueling floor. The SG primary assemblies would be loaded into shipping packages (large diameter steel pipes with welded bottom plates) situated vertically, and steel lids would be bolted onto a flange near the top of the package. The use of poly film or Herculite, which would function to minimize the spread of contamination, could be eliminated if the modules are packaged on the PCRV ledge. In addition, DP Section 3.3.3.4 states that it is anticipated that the steam generators will be shipped by rail for burial. The current plan is to ship the steam generators by truck over public highways.

The planned shipping container will be air tight and will meet the regulatory requirements for shipping and burial. The container will be supported in a vertical position on the ledge inside the PCRV or on the refueling floor, so that it cannot topple when the SG primary assembly is loaded into the container and the crane released from the assembly to enable the lid to be bolted onto the container. The shipping container currently planned will have 1/2 inch thick steel walls with little clearance between the walls and the SG primary assembly. Calculations, based on surveillance of SG primary assemblies, indicate the steel container would provide adequate shielding of a SG primary assembly without the addition of grout. Provisions exist for adding one inch thick steel side walls to the shipping trailer, if additional shielding is necessary. The total weight will be low enough to permit shipment by public highway rather than by rail.

The probability of dropping a SG primary assembly (evaluated in DP Section 3.4.10) is not increased. The container will be adequately supported when the crane is not connected to it, and the weight of the SG primary assembly, shipping container, and absorbent material is estimated to be less than 80,000 lbs., within the rating of the Reactor Building crane's 50 ton hook that will be used to lower the loaded container down the truck bay. The total time the container is suspended, susceptible to a potential drop accident, is essentially unchanged from the methodology evaluated in the DP. The consequences of a SG primary assembly drop accident are unchanged since the total activity assumed available for release is no different than that evaluated in DP Section 3.4.10. It was determined that no new accidents or malfunctions would be created by the alternative SG assembly handling and packaging, since the proposed new shipping container will meet regulatory requirements for transportation of radioactive materials and the container will be handled with the Reactor Building crane. The operations involved in the alternative handling and packaging do not reduce any margins of safety in the bases for Technical Specifications.

Based on the above, it was concluded that the alternative method for handling and packaging SG primary assemblies does not constitute an unreviewed safety question.

REFERENCE

1. "Supplement to Applicant's Environmental Report, Post Operating License Stage, for Proposed Decommissioning of the Fort St. Vrain Nuclear Generating Station"; Submitted to the NRC by PSC letter, Warembourg to Director, Office of Nuclear Reactor Regulation, dated April 30, 1992 (P-92181).