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March 4, 1993
Fort St. Vrain
Unit No. 1
P-93012

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

Docket No. 50-267

Subject: QUARTERLY SUBMITTAL OF THE REPORT OF CHANGES, TESTS AND EXPERIMENTS FOR FORT ST. VRAIN DECOMMISSIONING

References: 1. Facility Operating License No. DPR-34
2. NRC Letter dated November 23, 1992, Erickson to Crawford (G-92244)

Gentlemen:

This letter transmits the first Quarterly Report of Changes, Tests, and Experiments affecting Decommissioning of the Fort St. Vrain (FSV) Nuclear Station. The attached report includes a brief description of each change, test and experiment as well as a summary of the safety evaluation. This report covers the period of November 15, 1992 through February 15, 1993.

This report is being submitted pursuant to Condition (b)(2) of the "Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility", transmitted in Reference 2, 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."

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/jrj

Attachment

080115

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

Regional Administrator, Region IV

Mr. Ramon E. Hall, Director
Uranium Recovery Field Office

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

MARCH 1993
QUARTERLY REPORT OF CHANGES, TESTS AND EXPERIMENTS
FOR FSV DECOMMISSIONING

Background:

The following is a brief discussion of 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 15, 1992 to February 15, 1993. The FSV Decommissioning Order was issued on November 23, 1992. It should be noted that several of the items discussed in this report relate to the NRC's Safety Evaluation and Environmental Assessement for decommissioning that accompanied the Decommissioning Order, issued in NRC letter dated November 23, 1992, Erickson to Crawford (G-92244). These items represent inconsistencies between the NRC's Safety Evaluation and Environmental Assessment and the licensing basis for FSV decommissioning as described in the DP and Decommissioning Technical Specifications. PSC brought these inconsistencies to the attention of the NRC in a phone conversation dated November 25, 1992. The NRC recommended that PSC include an explanation of these items in the first quarterly report of changes, tests and experiments to prevent future misunderstandings.

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, transmitted in Reference 1, which states:

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

<u>Inconsistencies Between the Decommissioning Plan and the Safety Evaluation/Environmental Assessment that accompanied the FSV Decommissioning Order</u>

1. Processing of Leakage from the PCRV Shield Water System

Page 3 of the NRC's Safety Evaluation states that all leakage resulting from flooding the PCRV will be treated by means of the disposal demineralization and filtration system that is part of the PCRV Shield Water system. DP Section 2.3.3.6.4 states that major components of the system will be prefabricated on skids with drip pans to contain potential leakage, but does not discuss how the

leakage will be processed. Provisions are included in the system design to collect leakage from skid mounted components and direct this leakage to the existing Radioactive Liquid Waste System (System 62), where the liquid will be processed (including filtration and demineralization as necessary) and monitored prior to release. This system was used to process and monitor radioactive liquids during FSV's operational and defueling phases, and will continue to be used during decommissioning, as described in Sections 2.3.3.6 and 3.3.2 of the DP. Releases from this system shall be in accordance with the requirements of the FSV Offsite Dose Calculation Manual (ODCM), which assures compliance with 10 CFR 20 and 10 CFR 50 Appendix I.

Leakage from other parts of the PCRV Shield Water System that are not skid mounted would enter the Reactor Building sump (System 72). Releases from the Reactor Building sump are treated as radioactive releases, and are permitted only after analysis of a sample of the contents of the sump to determine activity concentrations and corresponding permissible release rates and dilution flows in accordance with the ODCM.

Evaluation determined that this method of handling leakage from the PCRV Shield Water System was acceptable and did not involve an unreviewed safety question. System 62 and System 72 were used to process and monitor radioactive liquids during FSV's operational and defueling phases, and their use during decommissioning will not impose any new performance requirements on these systems such that any design criteria is expected to be exceeded. Processing leakage from the Shield Water System in either System 62 or 72 prior to release does not affect the probability of occurrence of an accident or introduce new accidents. DP Section 3.4.7 evaluates the conseqences of a major breach of the PCRV Shield Water System, resulting in the entire shield water inventory draining into the Reactor Building basement. Projected consequences of this accident are based on a conservative evaporation rate of tritiated water and an extremely high tritium concentration, limited by Decommissioning Technical Specification 3.4. Should such an event occur, PSC could pump the water back into the PCRV Shield Water System for processing, transfer the water to the Radioactive Liquid Waste System for processing, or release directly from the reactor building sump, depending on activity levels, in accordance with the ODCM. The inconsistencies in descriptions of the PCRV Shield Water System described herein will not affect the consequences of this accident and the margin of safety in the basis for LC 3.4 would not be reduced.

2. **Location of the Decommissioning Exclusion Area Boundary**

Page 6 of the NRC's Safety Evaluation states that an area approximately 1 mi^2 within the site area is designated as the Exclusion Area and the distance from the Reactor Building to the

nearest Exclusion Area Boundary is about 1935 feet. This information corresponds to FSV site information in Section 2.2 of the Updated FSAR and Section 6.3 of the Technical Specifications which were in effect prior to the start of decommissioning. PSC has revised the Exclusion Area Boundary (EAB). The EAB that is in effect for decommissioning is shown in Figure 4-1 of the Decommissioning Technical Specifications, and corresponds to the Emergency Planning Zone (EPZ) boundary. The Exclusion Area covers an area of approximately one-tenth mi^2, and the closest distance from the Reactor Building to the EAB/EPZ boundary is 525 feet (160 meters). The accident analyses in the DP conservatively assume that an individual is stationed at a distance of 100 meters from the Reactor Building during accidents (well within the EAB/EPZ boundary), with the dispersion factor based on 100 meters for dose calculation purposes. Decommissioning accident doses, identified in Section 3.4 of the DP, are orders of magnitude below the 10 CFR 100 guidelines and a small fraction of the one Rem whole body dose and five Rem to any specific organ guidelines cited in the EPA Protective Action Guidelines (Reference 2). It was determined that the condensed Exclusion Area is consistent with decommissioning accident analyses, is easier for PSC to maintain control over than the previous larger area, and does not involve an unreviewed safety question.

3. **Absence of Charcoal Adsorbers in the Reactor Building Ventilation Exhaust System**

Page 9 of the NRC's Environmental Assessment that accompanied the Decommissioning Order stated that dust generated by the concrete cutting operations would be "filtered by the existing reactor building ventilation system consisting of moisture separators, HEPA filters and charcoal absorbers." The NRC concluded that dust generated during decommissioning would not impose a significant environmental impact. The charcoal sections have been removed from the Reactor Building ventilation exhaust system since they constituted a fire hazard and were no longer necessary. The purpose of the charcoal was to remove radioactive iodine from the exhaust stream in the event of an accident involving release of significant quantities of fission products. However, I-131, with an 8 day half-life, has decayed away. Other radioactive iodine isotopes, such as I-129, have been below the lower limits of detection in the gas inside the PCRV. The decommissioning accident analyses take no credit for filtration of gaseous effluent by the charcoal sections, and the DP does not refer to the charcoal. While the Decommissioning Technical Specifications requires operability of the HEPA filters in the Reactor Building ventilation exhaust system, there are no requirements pertaining to the charcoal sections. Therefore, it was determined that absence of the charcoal does not constitute an unreviewed safety question.

From an environmental standpoint, the only concern associated with

removal of the charcoal is the ability to filter dust, which could be produced by some of the decommissioning activities. Excellent dust filtration is provided by the HEPA filters in the Reactor Building exhaust system. These filters, described in Section 2.2.3.13 of the DP, have a specified and tested capability for removal of 99.9% of particulates that are 0.3 microns and greater in size. This filtration capability will serve to remove dust from the air before it exits the Reactor Building, and the charcoal sections are not necessary for dust removal. Therefore, it was determined that absence of the charcoal sections does not involve an unreviewed environmental question.

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

1. **Changes in the PCRV Shield Water System Design**

Section 2.3.3.6.4 of the DP identifies two demineralizers in the PCRV Shield Water System, sized to process a minimum flow of 100 gpm. The final design of the Shield Water System calls for eight roughing and two polishing demineralizers. The roughing demineralizer flow path will be sized for a minimum process flow of 200 gpm. This section of the DP, and Figure 2.3-4, also identify a demineralizer surge tank, a demineralizer pump, and a demineralizer booster pump, all of which have been eliminated from the final design. By increasing the design flow rate and number of demineralizers, the Shield Water System performance has been enhanced. Calculations show that the 200 gpm flow through eight roughing demineralizers provides assurance that the dose rate to workers on the rotary work platform will be maintained below 2 mrem/hr, the guideline value identified in DP Figure 2.3-10. Additionally, hydraulic calculations show that the capacity of the main clarifying pumps will produce adequate flows throughout the Shield Water System, so that the demineralizer surge tank, demineralizer pump and demineralizer booster pump can be eliminated from the system design.

The demineralizer vessels will be designed, fabricated and tested in accordance with ASME Code Section VIII. Piping and valves associated with the demineralizers will be designed, fabricated and tested in accordance with ANSI B31.1, with welder qualification in accordance ASME Code Section IX. These requirements are in accordance with those established for the rest of the Shield Water System. Therefore, the probability of a leak of the Shield Water System is not increased. Section 3.4.7 of the DP evaluates the consequences of a loss of shield water, based on evaporation of tritiated water from the Reactor Building basement. This analysis conservatively assumes a tritium concentration of 62.4 μCi/ml, which is the limiting concentration established by Decommissioning Technical Specification 3.4. Since none of the system changes will affect tritium concentrations, the consequences of accidents

previously evaluated in the DP are not increased, margins of safety are not reduced and an unreviewed safety question is not created.

2. **Changes in the PCRV Shield Water System Chemistry Control**

Sections 2.3.3.6.4 and 2.3.3.6.5 of the DP identify a Shield Water System chemistry control program consisting of sodium hydroxide for pH control to minimize corrosion and maintenance of a constant hydrogen peroxide concentration to prevent biological fouling. The final design calls for use of Calgon LCS-20 for corrosion inhibition, with intermittent injections of hydrogen peroxide, as necessary, to prevent biological fouling. Calgon LCS-20 consists of a solution of sodium nitrite, sodium tetraborate and sodium hydroxide. Laboratory tests have proven that this additive will effectively inhibit corrosion of carbon steel so that water clarity can be maintained. The hydrogen peroxide will be added to the Shield Water System in batches as required to control biological fouling. This chemistry control method will be at least as effective as that described in the DP in preventing corrosion and maintaining visual clarity of the shield water. It was determined that this change in chemical treatment methods does not involve an unreviewed safety question.

3. **Change in Vent Path During Initial PCRV Filling Operation**

Sections 2.3.3.6.5 and 2.3.4.12 of the DP describe initial fill of the PCRV, and state that air and gas displaced by the water will be routed to the Radioactive Gas Waste System for storage and sampling prior to release. Current plans call for release of the displaced air directly to the Reactor Building exhaust, following sampling and analysis of the gas in the PCRV. An assessment was performed of gaseous release from the PCRV directly to the Reactor Building exhaust during initial fill of the PCRV to determine whether holdup of the gas by the gas waste tanks may be necessary to assure gaseous releases meet the requirements of the ODCM under worst case conditions. Tritium is the primary radionuclide of concern for gaseous releases since other radionuclides in the shield water are not expected to become airborne. The assessment concludes that there is a factor of 90 margin between the calculated tritium release rate and the ODCM allowable release rate assuming that the water in the PCRV contains the maximum tritium concentration, 62.4 μCi/ml, permitted by Section 3.4 of the Decommissioning Technical Specifications, and assuming 100% relative humidity of air in the PCRV. The dose to an individual assumed to be 100 meters from the Reactor Building for much of the fill operation, using these worst case assumptions, would be less than one mrem. The 62.4 μCi/ml limiting tritium concentration is based on the assumption that 100,000 Ci of tritium leaches out of the graphite, which is not considered credible. It is estimated that 500 Ci of tritium will leach out of graphite, resulting in a tritium concentration in the shield water of 0.4 μCi/ml. Expected gaseous release rates during

initial PCRV fill would result in tritium concentrations in unrestricted areas 10,000 times below 10 CFR 20 limits. Calculations also show that worst case tritium releases in gaseous effluents during initial fill would contribute only a small fraction of the tritium release allowed by 10 CFR 50 Appendix I. Administrative controls have been identified to govern PCRV gas sample frequency during the initial fill operation.

Routing displaced gas directly to the Reactor Building exhaust does not affect any accidents previously evaluated in the DP and has no effect on any margin of safety in the basis for a Technical Specification. This evolution will not cause tritium concentrations to approach 10 CFR 20 limits, even under worst case conditions. It was therefore determined that use of this vent path does not involve an unreviewed safety question.

4. **Removal of the Radioactive Gas Waste Compressors from Service**

Section 2.2.3.11 of the DP indicates that the gas waste compressors are normally in service. In fact, the gas waste compressors have been shut down and are no longer normally in service, since the PCRV is the only source of radioactive gas (tritium) and it is being vented directly to the Reactor Building ventilation exhaust system. The Radioactive Gas Waste System directs certain gas waste streams, which were normally radioactive during plant operations, to a vacuum tank. The gas waste compressors maintain the vacuum tank at a negative pressure and direct radioactive gases to one of two surge tanks, where radioactive gases can be sampled, analyzed and released at a controlled rate via the Reactor Building ventilation exhaust system. There are no longer any sources of radioactive gases that need to be collected in the gas waste vacuum tank. Given the existing tritium concentrations, it is not physically possible to release tritium from the PCRV fast enough to exceed the 10 CFR 20 maximum permissible concentration for unrestricted areas (all other nuclides in the PCRV are in concentrations that are below the lower limits of detection). Since it is anticipated that radionuclide concentrations will not increase prior to the introduction of water to the PCRV, the gas holdup capability of System 63 is not necessary.

It was determined that removal of the radioactive gas waste compressors from service does not involve an unreviewed safety question.

Tests or Experiments not Described in the Decommissioning Plan

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

REFERENCES

1. NRC Letter dated November 23, 1992, Erickson to Crawford (G-92244); Subject: "Order to Authorize Decommissioning of Fort St. Vrain and Amendment No. 85 to Possession Only License No. DPR-34 (TAC No. M82592)

2. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", EPA-520/1-75-001-A, U.S. Environmental Protection Agency, January 1990.