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December 21, 1995
Fort St. Vrain
P-95119

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 August 16, 1995 through November 15, 1995.

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,

Frederick J. Borst

Fredcrick J. Borst
Decommissioning Program Director

FJB/JRJ

Attachment

cc: Mr. Michael F. Weber, Chief
Decommissioning and Regulatory
Issues Branch

Regional Administrator, Region IV

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

DECEMBER 1995
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 August 16, 1995 through November 15, 1995.

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

1. Removal of Additional Beltline Concrete, Insulation and Liner in the Prestressed Concrete Reactor Vessel (PCRV) Lower Plenum

Based on evaluation of the results of the activation analysis, documented in Appendix II of the DP, it was originally concluded that removal of beltline concrete from the PCRV in the core area, extending from the top head liner at elevation 4864 ft. down to elevation 4821 ft. (several feet below the bottom of the core support floor), would be sufficient to meet final survey release criteria. Following removal of this beltline concrete, samples were taken of PCRV insulation cover plates, PCRV liner plate and PCRV concrete in the lower plenum area. Analysis of these samples indicated that additional beltline concrete needed to be removed from the PCRV lower plenum to meet release criteria. It was decided to remove an additional 26 feet of beltline concrete, from 4821 ft. to 4795 ft. elevations. While the upper beltline concrete segments were removed to an average depth of 31 inches, it was decided that a lesser depth of concrete removal was necessary in the lower plenum. Therefore, the lower plenum vertical back cuts will be made between every third tendon tube, in lieu of every tendon tube as was done for the upper beltline concrete segments, resulting in an average depth of 27 inches. Following 14 vertical radial cuts, the same as for the upper beltline segments, 14 concrete segments will be removed from the lower plenum, each approximately 26 ft. long by 8 ft. wide by 27 inches thick. The heaviest of these segments is estimated to weigh 37 tons, with lifts calculated for up to 40 tons, considering the weight of rigging components and water retained in the insulation. This is well within the capacity of the Reactor Building crane, using its 50 ton hook.

DP Section 2.3.3.5.2 documents the results of an analysis of PCRV structural integrity, considering removal of the upper beltline segments. A supplemental analysis of PCRV structural integrity was performed to factor in the proposed removal of additional beltline concrete in the lower plenum. This analysis concluded that the stresses in PCRV concrete and rebar following removal of the lower beltline concrete are within allowable limits and bounded by the results of the previous PCRV structural integrity analyses.

The probability of a concrete segment drop accident is not increased since the evolutions involved in physically rigging out the beltline sections are the same as those used for the upper beltline concrete, and the lower sections are within the capacity of the Reactor Building crane 50 ton hook. The evolutions involved with this concrete removal do nothing to increase the probability of crane failure. Since each of the lower concrete segments is projected to contain less than 1 millicurie of activity in the concrete, the consequences of a drop accident would be bounded by those identified in DP Section 3.4.3, "Dropping of Activated Concrete Segment Accident", which assumed 15.11 curies involved in the postulated drop of a PCRV top head concrete segment.

The supplemental PCRV structural analysis concluded that the PCRV will maintain its integrity for dead weight and seismic events during and following removal of the lower plenum beltline concrete. Therefore, no new accidents or malfunctions are created. Removal of the PCRV lower plenum beltline concrete segments does not affect the bases of any technical specification, and no margin of safety in the basis for any technical specification is reduced.

Based on the above, it was concluded that removal of the lower plenum beltline concrete does not constitute an unreviewed safety question.

2. Provide Access Point for Final Survey of Radioactive Liquid Effluent Discharge Piping

This change enables access to the inside of the radioactive liquid effluent (System 62) discharge piping, which exits the Reactor Building north wall approximately 5 feet below grade and is buried after it leaves the Reactor Building. This line is required to be radiologically surveyed as part of the final survey for site release. Engineering Change Request (ECR) No. 95-011 installs a "tee" to replace an elbow in the 3 inch diameter discharge line, prior to the line exiting the Reactor Building. A cap on the tee can be removed to permit access to the pipe interior, enabling insertion of survey instrumentation into the discharge pipe.

The 3-inch diameter tee is being installed in the Reactor Building in a location that is essentially the "high point" of the discharge portion of System 62. After penetrating the Reactor Building wall, the discharge line slopes downward to the oil/water separator. The top of the oil water separator is below the elevation of the planned tee, so water should not flow back to the tee from outside the Reactor Building. Inside the Reactor Building, the Reactor Building sump and sump pumps are located well below the elevation of the tee. Therefore, leakage of significant quantities of radioactive liquid is not expected during installation of the tee, and when the threaded cap is removed from the tee for discharge pipe survey operations.

A drain connection will be installed on the tee to permit drainage of any water prior to removing the threaded cap. Controls will be in place to collect leakage during tee installation and removal of the threaded cap. In addition, controls will be established to assure there are no radioactive liquid effluent releases during tee installation and when the threaded cap is removed from the tee. In the unlikely event of a breakdown of controls and assuming a radioactive liquid effluent release is made while the threaded cap is removed, the maximum volume of water that could leak at any one time is the amount of water in the Reactor Building Sump (RBS). Since this quantity is much less than the 423,500 gallons of water assumed in the Loss of PCRV Shielding Water Accident, evaluated in DP Section 3.4.7, and the tritium concentrations would be much lower than those assumed, the consequences of such an accident would be bounded by those previously analyzed. The tee and cap on the tee installed by ECR-95-011 have a pressure rating in excess of the discharge pressure of the RBS pumps, so the probability of an accident or malfunction involving a spill of radioactive liquids is not increased. Any water that were to leak from the tee would drain back into the RBS, with no uncontrolled release to the environment.

All radioactive liquid effluent releases will continue to use the same effluent release path and be procedurally controlled to prohibit the accidental discharge of radioactive liquids. As stated above, when the cap is removed from the tee, controls will be in place to prohibit radioactive liquid effluent releases, thus preventing a localized spill. Therefore, no new accidents or malfunctions are created. Requirements for radioactive effluent releases are contained in the Offsite Dose Calculation Manual (ODCM), which is referenced by the Decommissioning Technical Specifications. The margins established in the ODCM to assure regulatory compliance are unaffected by this modification, and no margins of safety are reduced in the basis of any technical specification.

Based on the above, it was concluded that installation of the tee enabling access to the inside of the radioactive liquid effluent discharge piping does not constitute an unreviewed safety question.

3. Changes to the FSV Final Survey Plan (FSP)

The NRC approved the FSV Final Survey Plan for Site Release (FSP), as modified by PSCo responses to NRC requests for additional information (RAI), on January 26, 1995 (Reference 1). PSCo provided the updated FSP in Reference 2, which incorporated these RAI responses into the FSP. This approved FSP supersedes and replaces Section 4 of the DP, "Final Radiation Survey Plan". The preface to the updated FSP contains criteria that permit implementation of certain revisions to the FSP without prior NRC approval. A safety evaluation was performed on recently proposed FSP revisions which determined these criteria were met, as discussed below.

FSP Section 3.5.1.g, "Chain of Custody", was revised to provide more specific and manageable requirements for samples while on site. Previously samples had to be maintained in the possession of the individual, under direct surveillance, or secured. The FSP is revised to allow labeling, control, and storage of samples so as to maintain positive control of samples throughout the process. The change in sample chain of custody requirements places the reliance of sample control on a systematic approach as opposed to relying on individual possession. This logic is consistent with NUREG/CR-5849 Section 4.1.1, "Sample Chain-of-Custody."

FSP Section 3.5.1.1, "Software Control", was added for the validation, verification, and control of internal use final survey software. This is considered an enhancement to the control of final survey data.

FSP Sections 3.8.2, "Initial Classification", and 4.3.1, "Classification", were revised to elaborate on the bases for initial classification of survey units. This elaboration includes using characterization survey data, survey unit history, and potential for contamination. Previously, the FSP stated use of the potential for residual contamination as the basis for initial classification. This is considered an enhancement to the initial classification requirements.

FSP Section 3.8.11, "Investigation", was revised to state when and how investigations will be performed. These investigations will be implemented when survey measurements exceed the action levels. The intent of these investigations is to identify the causes for measurements in excess of action levels and to identify instances where licensed material is the cause for measurements exceeding action levels. Additionally, for plant system survey units, measurements at adjacent locations and biased locations may be required as a part of this revision. The change in investigation requirements is tied to the action levels. The previous discussion was not explicit enough as to the investigation requirements. The inclusion of surveying adjacent and biased locations for plant system survey units enhances the investigation process.

FSP Section 3.8.12, "Reclassification", was revised to state that in addition to reclassifying a survey unit, it may require resurveying also. A discussion was added explaining the reclassification requirements where licensed material is the cause of the elevated measurement results. This is considered an enhancement to the reclassification process.

FSP Section 3.9, "Schedule", was revised to indicate the use of groupings, as opposed to phases, and to indicate the use of release records. This reflects the current plans for managing the final survey process and is an administrative change to the FSP.

FSP Section 4.3.3, "Measurement Frequency" subsection "Affected Open Land Area", was revised to eliminate the discussion of the effluent pathway, since this area will be surveyed as described for affected open land areas. The previous discussion of the effluent pathway in the measurement frequency section of the FSP provided a general discussion of taking samples, but did not provide measurement frequencies.

FSP Section 6.1, "Topical Outline", was revised to indicate the use of groupings, as opposed to phases, and provides the current outline for the final survey reports. This is an administrative change to the FSP.

FSP Section 8.0, "Glossary", had various definitions revised based on the above changes.

It was determined that the FSP revisions described above do not constitute an unreviewed safety question and meet the following additional criteria defined in the FSP that authorize PSC to make changes to the FSP without prior NRC approval:

- The proposed revisions do not require changes to the Decommissioning Technical Specifications.
- The proposed revisions do not reduce the required survey frequency for the classification of a survey unit.
- The proposed revisions do not increase the action levels for conducting investigation and followup surveys.
- The proposed revisions do not affect the statistical

treatment of survey data in a manner which could reduce the confidence that the site meets the criteria for unrestricted use.

Based on the above, it was concluded that the FSP revisions do not constitute an unreviewed safety question, and may be implemented without prior NRC approval in accordance with the FSP requirements.

Tests or Experiments Not Described in the Decommissioning Plan

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

References

1. NRC letter, Pittiglio to Crawford, dated January 26, 1995 (G-95020)

2. PSC letter, Fisher to Weber (NRC), dated May 25, 1995 (P-95050)