PUBLIC SERVICE COMPANY OF COLORADO

FORT SAINT VRAIN STATION

ANNUAL REPORT OF CHANGES, TESTS, AND EXPERIMENTS NOT REQUIRING PRIOR COMMISSION APPROVAL PURSUANT TO 10 CFR 50.59

January 23, 1991 through January 22, 1992

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INTRODUCTION

This report is submitted to comply with the requirements of Part 50.59(b) Title 10, Code of Federal Regulations, as they apply to Fort St. Vrain Station, Unit No. 1. It includes the period January 23, 1991 through January 22, 1992.

The following defines certain activities contained in this report:

Change Notice (CN) - A document containing installation, inspection and testing requirements, design background information, and design document updating requirements which specify the design control requirements applicable to a plant modification and authorizes changes to "as-built" plant design documentation.

Document Change Notice (DCN) - A document which authorizes a change to design documents. As a minimum, it contains a design input statement, a design analysis statement, a document update list and the document update information.

Setpoint Change Report (SCR) - A document which authorizes setpoint changes which do not constitute an alteration to the design of the affected equipment.

T-Tests - Special tests proposed and conducted by Public Service Company of Colorado.

The following is a list of standard abbreviations used at Fort St. Vrain:

AC - Alternating Current

ACM - Alternate Cooling Method

CRD - Control Rod Drive

DCCF - Document Change Coordination Form

EMF - Electro Motive Force

EQ - Environmental Qualification

FHM - Fuel Handling Machine

FPPP - Fire Protection Program Plan

FSAR - Final Safety Analysis Report

FSV - Fort St. Vrain

HELB - High Energy Line Break

HVAC - Heating, Ventilating, and Air Conditioning

LCO - Limiting Condition for Operation

ISFSI - Independent Spent Fuel Storage Installation

LER - Licensee Event Report

LTA - Low Temperature Adsorber

MCC - Motor Control Center

P&I - Piping and Instrument Drawing

PCRV - Prestressed Concrete Reactor Vessel

PPS - Plant Protective System

RERP - Radiological Emergency Response Plan

SOP - System Operating Procedure

SR - Surveillance Requirement

The following defines terms used in safety evaluation summaries contained in this report:

Enhanced Quality

Items for which quality program requirements have been identified, but which are not safety related. This includes non-safety related fire protection (System 45 excluding safety related portions), portions of the Independent Spent Fuel Storage Installation (ISFSI), Security (System 78 excluding Gai-Tronics), and packaging and transportation of radioactive materials.

Safety Related Items

Those plant systems, structures, equipment, and components which are identified by the FSAR, and as detailed and supplemented by applicable P&I, IB and IC diagrams; E and E-1203 schematic diagrams; the Cable Tab; SR-6-2 and SR-6-8 lists to include the following:

a) Class I per Table 1.4-1 of the FSAR.

b) Safe shutdown components per Table 1.4-2 of the FSAR.

c) Alternate Cooling Method (ACM) system.

<u>EXCEPTION</u>: The ACM system is exempt from requirements for seismic and environmental qualification.

d) Interface circuits (IC) within the Environmental Qualification Program.

<u>Safety Significant Change</u>

Changes to the facility, systems, components, or structures as described in the FSAR that may do any one (1) of the following:

a) Affect their capability to prevent or mitigate the consequences of accidents described in the FSAR.

b) Could result in exposures to the plant personnel in excess of occupational limits.

Changes in the safety related systems which involve the addition, deletion, or repair of components, structures, equipment, or systems such that the original design intent is changed (i.e., changes in redundancy, performance characteristics, separation, circuitry logic, control, margins of safety, safe shutdown, accident analysis, or any change that would result in an unreviewed safety question or require a Technical Specification change).

<u>Unreviewed Safety Question</u>

Any plant modification or activity that is deemed to involve an unreviewed safety question as defined in 10 CFR 50.59.

a) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

b) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created.

c) The margin of safety as defined in the basis for any Technical Specification is reduced.

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Background:

The following is a brief discussion of the changes, tests, and experiments affecting the Fort St. Vrain Station in the time period from January 23, 1991, to January 22, 1992, that have not been previously reported to the Nuclear Regulatory Commission (NRC). It should be noted that many of the activities discussed in this report are directly related to the permanent shutdown condition of the FSV reactor, the defueling and decommissioning preparations of the plant.

1.0 CHANGE NOTICES (CN)

CN-2213

System 79/Technical Support Center

CN-2213 was initiated to provide air monitoring capabilities in the Technical Support Center (TSC) utilizing an installed monitor, RIT-7937. Previously, room air in the TSC was monitored by a portable unit which produced high noise levels and was unreliable. RIT-79[illegible] can be aligned to sample either outside air drawn in by the HVAC system or room air within the TSC. The TSC room air monitoring is accomplished by manual actuation of solenoid valves in sample lines.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

CN-2854 and CN-2854A

System 52/Turbine Steam

CN-2854 changed manual valve V-5288, Bypass Flash Tank drain system valve, from a gate valve to a globe valve. The gate valve had been used as a throttling valve which resulted in excessive internal erosion. The globe valve was fully qualified for the intended service and better suited for the specific application. The modification was considered Enhanced Quality since V-5288 was a component of Appendix R Fire Protection Shutdown/Cooldown Train A. The change from a gate valve to a globe valve was evaluated and determined to have no affect on the heat removal rate of Train A.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

CN-2956 and CN-2956A

System 23/Helium Purification System

CN-2956 mechanically and electrically removed Helium Purification Regeneration components from the System 23 regeneration pit. The modification allows use of the pit as a loading port for defueling elements. As previously reported, CN-2953 structurally modified the pit for use as a loading port for defueling elements.

CN-2956A removed associated control and power supply cables from panels located in the Control Room, and cable trays located throughout the plant.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-2965 and CN-2965A

System 14/Fuel Storage Facility

CN-2965 provided an additional seismically qualified Reactor Isolation Valve (RIV) and a seismically qualified spacer to improve defueling operations. The spacer acts only as a seismically qualified permanent storage pedestal. The new RIV and spacer satisfy the requirements of Technical Specification LCO 4.7.2 in providing a seismically qualified location to store the Fuel Handling Machine when it is not attached to the Reactor Building overhead crane.

CN-2965A was issued to qualify the new RIV as safety related for use over the reactor vessel and fuel storage wells, and provide a document update package. The RIV, as originally purchased, was not supplied with safety related seals and/or the seal(s) failed leak tests as performed by the manufacturer. Qualified seals were placed on the RIV and successfully tested.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3009

System 84/Auxiliary Boiler and Heating System

CN-3009 was written to replace the existing backup auxiliary boiler (S-8401S) with a lower steam generating capacity boiler. The new backup auxiliary boiler has a rating of less than 15,000 lbm/hr and 650 degrees F. Meeting these capacity requirements ensured a harsh environment, as described in 10 CFR 50.49, could not be created in the event of a steam line rupture at Fort St. Vrain. During defueling of the FSV Reactor, a greater steam capacity was not needed and environmental qualification in accordance with 10 CFR 50.49 was not required.

The auxiliary boiler (S-8401) was down rated by CN-3006 to meet the environmental qualification requirements of 10 CFR 50.49. This activity was reported in the 10 CFR 50.59 report submitted to the NRC in 1991.

Replacement of the backup auxiliary boiler (S-8401S) was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3024

System 36/Independent Spent Fuel Storage Installation
System 83/Communication System

CN-3024 constructed a new asphalt road from the Reactor Building to the north-east corner of the Protected Area fence. The road provided a route to the Independent Spent Fuel Storage Installation (ISFSI) to allow transport of spent nuclear fuel to the ISFSI. Gates were installed in the inner isolation zone fence and the Protected Area fence. Communication cables were also installed for the in-plant system and the telephone system.

This activity was not safety related or safety significant, and did not involve an unreviewed safety [illegible]

CN-3037

System 11/Prestressed Concrete Reactor Vessel
System 93/Controls and Instrumentation

CN-3037 provided the design engineering and analyses to support the addition of the Startup Channel nuclear instrument inputs to the plant data logger. This required the addition of an acceptable means of isolation between safety related signals and non-safety related equipment to prevent system degradation for postulated failures.

The startup channels monitored core reactivity while the reactor was shut down and being defueled. Since the data logger system is non-safety related and the startup channel inputs were safety related, the isolators prevented a startup channel failure resulting from any postulated failure in the data logger system.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3038

System 21/Primary Coolant System

CN-3038 installed piping and valves between the Emergency Condensate Header and the Loop 2 Helium Circulator Bearing Water Surge Tank, T-2105. This provided a permanent water supply to the tank during shutdown conditions and reduced wear and maintenance on the Bearing Water Makeup Pumps. The modification closed out a temporary configuration which utilized a flexible hose.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3041

System 31/Feedwater and Condensate

CN-3041 positively isolated lines to prevent the entry of water and to assure a dry layup of the feedwater heaters. The feedwater heaters were equipment items not required to provide core cooling or support defueling operations, and for preservation purposes, were isolated, drained, and placed in dry layup. As is discussed in FSAR Section 14.14.2.7, reactor decay heat removal could be accomplished with the PCRV Liner Cooling System, System 46, only.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

CN-3046

System 11/Prestressed Concrete Reactor Vessel

CN-3046 was initiated to obtain approximately six samples of PCRV concrete. The samples were obtained by core drilling into the PCRV outer surface. The core drills were a nominal diameter of two inches and a nominal depth (length) of four inches. The samples were used to determine the concentration of impurities in the concrete to validate assumptions made in the neutron activation analysis, in support of decommissioning FSV. The samples were taken from elevations corresponding to various elevations of the active core. PCRV wall thickness in this area is approximately nine feet. The holes were refilled with grout of comparable compressive strength to that of the original PCRV concrete.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3050

System 11/Prestressed Concrete Reactor Vessel
System 93/Controls and Instrumentation

CN-3050 disabled the three Wide Range Channels (WRCs) of the nuclear instruments by placing modified bistable cards into the Plant Protective System (PPS) modules in the Control Room. The modification also disabled the Six Linear Power Channels. The two Startup Channels were not adversely affected by the modification.

The WRCs provided two protective functions: (1) a Rod Withdrawal Prohibit, and (2) reactor scram. With the FSV reactor permanently shut down, neutron flux well below the operating range of the WRCs, and with controls in place to prevent any approach to criticality during defueling, the protective actions associated with the WRCs were no longer necessary.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3051

System 46/Reactor Plant Cooling Water System
System 75/Turbine Building

CN-3051 installed piping and valves to collect condensate from the heating coils of the Loop 1 and Loop 2 System 46 surge tanks and deliver that condensate to either the Miscellaneous Drain Tank, T-7505, or the Drain Collector Tank, T-7507. The normal drain path was to the Reactor Building Sump and was retained as a backup. Benefits of the new drain path included condensate recovery and enhanced water quality.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3052

System 84/Auxiliary Boiler and Heating System

CN-3052 installed a new deaerator unit, complete with its own feedwater pumps, in the auxiliary boilers' feedwater system. Excessive boiler down time was believed to be due to high dissolved oxygen content in combination with low feedwater temperature. The auxiliary boiler(s) will continue to be needed throughout the decommissioning phase of Fort St. Vrain.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

CN-3058

System 22/Secondary Coolant System

CN-3058 removed the main steam ring header from steam generator module B-1-1 and portions of the main steam ring header from module B-1-2. Extensive cracking in the main steam ring headers of several of the 12 steam generator modules lead to the premature permanent shutdown of Fort St. Vrain in August, 1989. The ring headers were delivered to EG&G Idaho, for thorough metallurgical examination. The effort was directed at determining the root cause of the ring header failures in order to support the Department of Energy's interests in applying FSV experiences to any new HTGR design (i.e., technology transfer).

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

2.0 DOCUMENT CHANGE NOTICES (DCN)

None

3.0 SETPOINT CHANGE REPORTS (SCR)

SCR 91-002

System 33/Water Treatment

Following permanent shutdown of FSV in August, 1989, water chemistry limits were necessarily evaluated and changed. System temperatures and contaminant levels were quite different for the shutdown plant and the use of different chemicals was also considered. The steam/water combined chemistry sample recorder and alarm unit R-33200 was reset to alarm at the designated level and recorder points, based on the water chemistry unit's analysis of shutdown conditions and available chemicals to control shutdown plant chemistry parameters.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

SCR 92-001

System 46/Reactor Plant Cooling Water System

SCR 92-001 lowered the setpoints of TSL-4637 and TSL-4638. The two temperature switches provide a low temperature alarm for each of the two loops of the PCRV cooling water circuits. The alarms were set a few degrees (approximately 4 degrees F) above the low limit specified in Technical Specification (TS) LCO 4.2.15. Amendment No. 83 to the FSV TS approved lowering the temperature limits of LCO 4.2.15 based on the permanent shutdown and defueling condition of the reactor. SCR 92-001 implemented TS Amendment No. 83, which lowered the allowable operating temperature band by 15°F.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

4.0 SPECIAL TESTS (T-TESTS)

T-423

System 13/Fuel Handling Equipment

Purpose: T-423 was written to support the Engineering Evaluation Test Series that was performed in 1989 prior to defueling. The purpose of the test was to exercise each of the Fuel Handling Machine (FHM) drive systems and the grapple head sensors while monitoring and recording the analog and digital signals with a PC-based data acquisition system (DAS). The recorded signals were to provide baseline data against which the planned upgrades of the FHM could be judged for performance evaluation.

Results: The DAS has been used on numerous occasions and has been permanently installed by CN-3049. The DAS was also instrumental in making other upgrades associated with the FHM successful. This test proved the DAS, and the DAS enhanced the reliability of the FHM.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-441

System 13/Fuel Handling Equipment

Purpose: T-441 was written to assist in learning how to effectively use the FHM DAS It provided a means by which the process of determining how to make the physical connections to the FHM circuitry, implementing these plans, and performing the system monitoring desired could be extended in time without having to remove the connections and reconnect each day. T-441 was used for initial assessment of a problem before permanent connections were made by a design modification, or for general system monitoring.

Results: As with T-423 (previous page), the most critical connection points associated with T-441 have been selected and installed via CN-3049 for further enhancement of FHM operation and reliability.

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

T-454

System 11/Prestressed Concrete Reactor Vessel

Purpose: T-454 was initiated to remotely obtain and record the radiation dose rates at various elevations within a helium circulator penetration. Data collected will assist personnel involved with the decommissioning of Fort St. Vrain in the assessment for manned access to penetration primary flanges.

Results: T-454 data was not required to support any conclusions, but simply to allow habitability assessment. Elevations monitored were from approximately 4768 feet to approximately 4792 feet. Dose rates ranged from 0 mr/hr at approximately five feet into the penetration, up to 72 mr/hr at approximately 24 feet into the penetration and lower portion of the PCRV (i.e., 0 cpm to 1.6K - 2K cpm).

This activity was not safety related or safety significant, and did not involve an unreviewed safety question.

T-456

System 11/Prestressed Concrete Reactor Vessel

Purpose: T-456 was initiated to remotely access a Loop 1 steam generator module hot reheat pipe to obtain and record radiation dose rates at various elevations within the module. Data collected will assist decommissioning personnel in their assessment for manned access to various steam generator components and elevations.

Results: T-456 data was not required to support any conclusions, but simply to allow habitability assessment. Elevations monitored were from approximately 4768 feet to approximately 4815 feet. Dose rates ranged from less than 5 mr/hr at the first measurement point up to approximately 300 mr/hr at 42 feet into the steam generator module hot reheat pipe (i.e., 10 to 30 cpm to approximately 7.5K cpm).

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-457

System 21/Primary Coolant System

Purpose: T-457 was initiated to help determine operability of C-2103, the helium circulator 1C brake system, and potential cause(s) of the brake problem. The test was divided into two parts. Part 1 applied 1000 psig to the brake supply line which, with no motive power to drive the circulator, would normally stop it from self-turbining. Part 2 of the test actuated the pressurizing line several times and monitored the pressure drop in the supply bottle.

Results: Part 1 did not stop the circulator, indicating no brake pad contact with the circulator shaft. Part 2 indicated that three (3) actuations reduced bottle pressure 20 psi. When compared to actuating requirements for another helium circulator, the difference was negligible and indicated no internal blockage of the brake pressurizing line existed. At that time it was determined that only circulator removal and visual inspection of the brakes could resolve brake status.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-458

System 93/Controls and Instrumentation

Purpose: T-458 was written to determine the effects of permanent de-energization of Wide Range Channels (WRCs) III, IV, and V. During reactor operations prior to beginning defueling, the WRCs provided two functions: (1) Rod Withdrawal Prohibit, and (2) reactor scram. The interlock/trip functions related to the WRCs were not required during defueling, and only served to prolong defueling by unnecessary actuations of the Plant Protective System.

Results: The test was successfully performed with all requirements met and no abnormal conditions observed. CN-3050 (this report) permanently disabled the WRCs.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-459

System 13/Fuel Handling Equipment

Purpose: T-459 was written to gather preliminary data concerning loose surface contamination that would be created during transfer of irradiated fuel elements into an Independent Spent Fuel Storage Installation (ISFSI) fuel storage canister. The contributing components were the Reactor Isolation Valves' (RIVs) gates during opening and closing, and the fuel elements themselves. Based on the design of the ISFSI, significant loose surface contamination is not acceptable on a fuel storage canister during storage.

Results: Although contamination levels were less than anticipated, they were still higher than allowable limits specified in the ISFSI Technical Specifications. The test showed that the RIV used did not contribute any detectable amount of contamination during normal operations. However, loose surface contamination on the top of a fuel storage canister from actual fuel element movements was a concern. T-459 showed that further testing was needed and decontamination of the ISFSI fuel storage canisters used for the test would not be difficult.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-460

System 13/Fuel Handling Equipment

Purpose: T-460 was written to determine the best method(s) of mitigating loose surface contamination caused by handling irradiated fuel elements (see T-459 previous page) in preparation for defueling the FSV reactor to the ISFSI.

Results: T-460 utilized several mitigation methods in the control of loose surface contamination. The best results were obtained using the maslin wrapped ISFSI spacer ring test. The ISFSI spacer is used between a RIV and the fuel storage canister. This leaves a gap of about 0.080 of an inch between the outside diameter of the spacer ring and the inside diameter of the lower portion of the RIV seal ring. During fuel transfer, loose surface contamination could pass through the gap and contaminate the outer surface of the fuel storage canister. The maslin wrap was effective in mitigating this source of contamination to the outside of the fuel storage canister.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-461

System 46/Reactor Plant Cooling Water System

Purpose: T-461 was initiated to determine the effects of a reduced water flow in one loop of System 46 with the other loop shut down. Flows were reduced to approximately 25% of normal in the PCRV lower barrel, bottom head, and core support floor areas while the upper barrel and top head flows were reduced to approximately 75% of normal. During defueling of the FSV reactor, decay heat generation was low and in order to maintain System 46 water temperature within Technical Specification LCO 4.2.15 limits (i.e., 100 to 120 degrees F) auxiliary heating had to be provided. The amount of auxiliary heat required was very small and, therefore, not easily compatible with the heat source. The intent was to be able to shut down the auxiliary heat source for a financial and manpower savings.

Results: Typically, PCRV liner cooling water temperatures rose approximately one degree as measured by each subheader outlet thermocouple. A few tubes in the upper barrel and top head area [illegible] approximately four to five degrees F above subheader inlet water temperature. In summary:

1. Changes in System 46 flow and temperature did not appear to have a significant affect on liner cooling tube differential temperatures at the flow rates used in this test.

2. There was some heatup of the liner cooling tubes in the top head and upper barrel areas.

3. Reactor coolant flow and core temperatures had little, if any effect on liner cooling tube temperatures.

4. Reactor building ambient temperature had a significant effect on operating liner cooling tube temperatures.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

T-462

System 84/Auxiliary Boiler and Heating System
System 92/Accessory Electrical Equipment

Purpose: T-462 was written and performed to verify that three underground storage tanks at Fort St. Vrain were not leaking and did not present any potential for product (diesel fuel) release to the environment. The test utilized a vacuum technique whereby a slight vacuum was pulled on each tank tested. Sensitive instrumentation was used to monitor for bubble formation (indicating air ingress), or a change in water level (indicating water ingress).

Results: Testing indicated no apparent leakage in any of the three tanks.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.

5.0 PROCEDURES

DSOP 84-02, Issue 2

System 84/Auxiliary Boiler and Heating

Following completion of CN-3052, which installed a new deaerator unit for the auxiliary boilers, DSOP 84-02 was revised. This procedure provides detailed instructions for startup, operation, and shutdown of the auxiliary boilers. DSOP 84-02 also identifies appropriate precautions necessary to limit operations to one boiler at a time, and steam flow and temperature requirements within the limits established for the FSV environmental qualification program (10 CFR 50.49). Refer to CN-3009 and CN-3052 for additional information related to the auxiliary boilers.

This activity was classified safety related, but was not safety significant and did not involve an unreviewed safety question.