UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

PUBLIC SERVICE COMPANY OF COLORADO

DOCKET 50-267

FORT ST. VRAIN NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License DPR-34

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Public Service Company of Colorado (the licensee) dated August 23, 1984, as supplemented by letter dated October 12, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License DPR-34 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric H. Johnson

Eric H. Johnson, Chief
Reactor Project Branch 1

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 9, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE DPR-34

DOCKET 50-267

Replace the following pages of the Appendix A Technical Specifications with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
v	v
--	5.3-22 (new page)
--	5.3-23 (new page)

PAGE

5.2 PRIMARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS (Continued)

5.2.26 - Region Constraint Devices 5.2-36
5.2.27 - Helium Shutoff Valves 5.2-38
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5.3 SECONDARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS 5.3-1

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Specification SR 5.3.2 - Main and Hot Reheat Steam Stop Check Valves 5.3-3
Specification SR 5.3.3 - Bypass and Pressure Relief Valves 5.3-4
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Specification SR 5.3.5 - Hydraulic Power System 5.3-7
Specification SR 5.3.6 - Instrument Air System 5.3-8
Specification SR 5.3.7 - Secondary Coolant Activity 5.3-8
Specification SR 5.3.8 - Shock Suppressors (Snubbers) 5.3-9
Specification SR 5.3.9 - Safety Valves 5.3-18
Specification SR 5.3.10 - Secondary Coolant System Instrumentation 5.3-19
Specification SR 5.3.11 - Steam Generator Bimetallic Welds 5.3-20
Specification SR 5.3.12 - Steam Generator Tube Leaks 5.3-22

5.4 INSTRUMENTATION AND CONTROL SYSTEMS - SURVEILLANCE REQUIREMENTS .. 5.4-1

Specification SR 5.4.1 - Reactor Protective System and Other Critical Instrumentation and Control Checks, Calibrations and Tests 5.4-1
Specification SR 5.4.2 - Control Room Smoke Detector 5.4-12
Specification SR 5.4.3 - Core Region Outlet Temperature Instrumentation 5.4-12
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Specification SR 5.4.10 - Seismic Instrumentation 5.4-15
Specification SR 5.4.11 - PCRV Surface Temperature Indication 5.4-16
Specification SR 5.4.12 - Analytical System Primary Coolant Moisture Instrumentation 5.4-16
Specification SR 5.4.13 - 480 V Switchgear Room Temperature Indication 5.4-16

Specification SR 5.3.12 - Steam Generator Tube Leaks Surveillance

Following each steam generator tube leak, specimens from the accessible subheader tubes connected to the leaking inaccessible tube(s) shall be metallographically examined. The results of this examination shall be compared to the results from the specimens of all previous tube leaks.

A study shall be performed to evaluate the size and elevation of all tube leaks to determine if a cause or trend in the degradation of the tubes can be identified.

Following each steam generator tube leak study, the Nuclear Regulatory Commission shall be notified as to the estimated size and elevation of the leaks as well as the results of the metallographic and engineering analyses performed that may identify the mechanism that caused the leak to occur.

Basis for Specification SR 5.3.12

During the life time of the plant a certain number of steam generator tube leaks are expected to occur, and the steam generators have been designed to have these leaking tube subheaders plugged without affecting the plant's performance as shown in FSAR Table 4.2-5. The consequences of steam generator tube leaks have been analyzed in FSAR Section 14.5.

It is important to identify the size and elevation of steam generator tube leaks and to metallographically examine the subheader tube material because this information can be used to identify any trend or generic cause of tube leaks. Conclusive identification of the cause of a steam generator tube leak will enable modifications and/or changes in operation to increase the reliability and life of the steam generators and to prevent tube failures in excess of those analyzed in the FSAR.

Because of the subheader designs leading to the steam generator's tube bundles, in-situ internal or external inspection and evaluation of a tube leak to establish a conclusive cause is not practical. Metallographic examination of the accessible connecting subheader tube will show the condition of the internal subheader wall, giving an indication of the conditions of the leaking tube internal wall, thereby demonstrating the effectiveness of water chemistry controls. Determining the exact size and elevation of the tube leak will enable evaluation of other possible leak causes such as tube/tube support plate interface effects.

The surveillance plan outlined above is considered adequate to evaluate steam generator tube integrity and assure that the consequences of postulated tube leaks remain within the limits analyzed in the FSAR.