UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO FORT ST. VRAIN STEAM GENERIC BIMETALLIC WELDS

FORT ST. VRAIN NUCLEAR GENERATING STATION

PUBLIC SERVICE COMPANY OF COLORADO

DOCKET NO. 50-267

1.0 INTRODUCTION

On March 17, 1986, a steam generator (boiler) tube failed at one of the Advanced Gas-Cooled Reactors (AGR's) at Hartlepool in the United Kingdom. This event was of immediate interest to the staff because of the similarity of the Hartlepool AGR steam generator design to that at Fort St. Vrain (FSV).

The FSV steam generators each consist of 6 modules. The gas coolant flows down through the modules, first over the reheater tubes, then over the main steam tubes. The tube bundles are helically wound. The main steam section tubes are fabricated from multiple materials, each tailored for the specific operating temperature range. The steam lines exit through penetrations in the bottom of the reactor vessel.

The AGR steam generators are similar in several major respects. The gas coolant also flows down through the steam generator, flowing first over a separate reheater tube bundle. The tube bundles are also helically wound. The main steam section tubes are fabricated from multiple materials. One major difference is that the steam lines are connected externally through the top of the reactor vessel.

Table 1 (attached) summarizes the temperatures and pressures of the reactor (primary) and secondary coolant. Except for the slightly more severe conditions in the FSV reactor (primary) coolant, secondary coolant conditions are very similar.

Subsequently, proprietary information was received from the OECD Nuclear Energy Agency, Incident Reporting System (IRS). The specific IRS reports are numbers 07.92.10 and 07.92.20. These reports concerned superheater transition joint cracking, found both at Hartlepool and Heysham 1. The staff's evaluation is based on this information, and knowledge about the construction of the FSV steam generators.

2.0 EVALUATION

2.1 Metallurgy

The staff's evaluation focused on the welding techniques used to join dissimilar tubing materials in the AGR and FSV steam generators.

The staff has reviewed the welding procedures for making the bimetallic transition welds in the steam generators at the Hartlepool and Heysham AGRs and at the FSV reactor. Inspection of the affected area in the AGRs showed that cracks were formed in the bimetallic transition weld in a brittle martensitic zone between the 2.25% Cr-1% Mo and Inconel materials.

The procedure used to fabricate the Hartlepool and Heysham bimetallic welds involved buttering the 2.25% Cr-1% Mo pipe with Inconel 182 and stress relieving (700°± 15°C. for 3 h.) the buttered welds in the shop. After pressure testing, the buttered pipe was transferred to the site and welded to the Type 316 stainless steel steam generator nozzle, using Inconel 182 filler metal. Neither a preheat nor a stress relief heat treatment was carried out on the welds on site. This procedure would account for the brittle martensitic zone observed at the junction of the 2.25% Cr-1% Mo and Inconel materials.

The procedure used to fabricate the FSV steam generator bimetallic welds was in compliance to Sterns-Roger Welding Procedure Specifications WS-5, Specification No. 205. Welder qualification tests were made and documented in accordance with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code for Class A pressure vessels and certified by the Hartford Steam Boiler Inspection and Insurance Company.

The Sterns-Roger specifications required that the weld be slowly cooled from the welding temperature and that the 2.25% Cr-1% Mo (P-5) material be preheated prior to welding to a minimum of 300°F. The filler material was to conform to ASME material specification SB-304, F-43 (ER NiCr-3). Preheat was not specified for the P-45 (33%Ni 21%Cr 43%Fe) material. This procedure ensured that a brittle martensitic zone would not form on the 2.25% Cr-1% Mo surface during welding of the steam generators at the Fort St. Vrain reactor. The anticipated life of the bimetallic welds prepared to the Sterns-Roger specifications exceed 1 x 10^5 hours.

2.2 System Safety

The staff also examined the metallurgical findings relative to the actual Fort St. Vrain operational experience. As of December 31, 1988, Fort St. Vrain had accumulated 37,245 hours of critical reactor operation and 25,073 hours of generator on-line operation. The latter figure more closely represents actual hours of operation of the steam generators at rated temperature and pressure conditions. At the present time, the steam generators are operating well within their design lifetime. A more extensive consideration of this issue would be required if anticipated plant operation actually approached the estimated bimetallic weld lifetime given above.

3.0 CONCLUSIONS

The staff finds that the steam generator tube failures experienced by the AGR's in the United Kingdom do not affect near term operation of Fort St. Vrain. The estimated life of Fort St. Vrain's bimetallic welds greatly exceeds actual service life. Therefore, the staff concludes that continued operation of Fort St. Vrain is not affected by the steam generator tube failures experienced by the AGR's.

4.0 REFERENCES

1. The mechanical design and validation of the helical tube boilers for Hartlepool and Heysham AGR stations. No. 32, "Gas-Cooled reactors today" BNES, London, 1983.

2. AGR boiler materials selection considerations No. 109, "Gas-cooled reactors today," BNES, London, 1983.

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Attachment:
Table 1

TABLE 1

COMPARISON OF TEMPERATURES AND PRESSURES

FORT ST. VRAIN STEAM GENERATORS VS HARTLEPOOL AGR BOILERS

Gas - Reactor (Primary) Coolant	FSV	AGR (Hartlepool)
Maximum Temperature, °F	1427	1189
Maximum Pressure, psia	700	573
Steam - Secondary Coolant		
Main Steam		
Maximum Temperature, °F	1000	1009
Maximum Pressure, psia	2400	2484
Reheat System		
Maximum Temperature, °F	1000	1002
Maximum Pressure, psia	600	602