



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

Enclosure 1

SAFETY EVALUATION  
OF PUBLIC SERVICE COMPANY OF COLORADO REPORT  
"SUPERHEATER TUBE LEAKS IN THE STEAM GENERATORS  
OF THE FORT ST. VRAIN HIGH TEMPERATURE  
GAS COOLED REACTOR" JANUARY 1984

Background

On December 8, 1982, a secondary side to primary side leak was discovered in the economizer-evaporator-superheater section of the B-2-3 module in the Loop 2 steam generator of the Fort St. Vrain nuclear plant. The leak was assumed to have developed following a reactor scram transient which occurred on September 30, 1982.

The leak elevation was located and the leaking tube was identified as one of three tubes connected to subheader "M" in the affected module. Based on leak rate results the hole (leak) was on the order of a 0.003 inch diameter orifice. The plugging operation involved the removal of the 3 tubes connected to subheader "M" from service out of 54 tubes in the affected module. In the plugging operation, sections from the feedwater lead in and the steam lead out tubes were removed and both ends of each tube were capped. A section of the steam generator tube, alloy 800 grade 1 and a section of the feedwater tube, carbon steel SA 210 type A-1 which were removed to perform the plugging operation were sent to General Atomics for laboratory examinations.

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In their report, dated January 1983 and entitled "Metallurgical Examination of Tubes Removed from Fort St. Vrain Steam Generator B-2-3", General Atomics presented the results of visual examinations and metallurgical examinations of the steam generator and feedwater tube sections.

Visual examination of the alloy 800 steam generator tube revealed an apparent thin oxide film on the exterior (gas side) and a thin coating on the inside of the tube; there was no evidence of corrosive attack. The feedwater tube section had uniform corrosion, as anticipated, with no evidence of anomalous degradation.

Metallurgical evaluation included metallographic mounting of specimens for microstructural examination, microhardness measurements and energy dispersive analysis (EDAX) for determining the composition of the oxide or corrosion films.

The oxide film on the alloy 800 steam generator tube consisted primarily of Fe-Cr-Ni oxide and had an average thickness of 0.008 inch with no microscopic evidence of pitting, cracking or erosion/corrosion damage. The microstructure was fine grained with evidence of cold work, primarily in bend sections but microhardness measurements did not suggest any extensive work hardening. At 1000 X magnification, the microstructure was considered typical for alloy 800 grade 1 and the grain boundaries were observed to be free of significant carbides precipitation indicating no degree of sensitization.

The feedwater tube corrosion film was magnetite with thicknesses ranging from 0.010 to 0.040 inch and averaging 0.021 inch (the tube wall thickness is 0.165 inch minimum). The EDAX analysis of the magnetite indicated iron as  $Fe_3 O_4$  and some silicon and copper were also detected. The presence of copper with some oxygen and chlorides in the system suggests the reason for the thick magnetite growth on the feedwater tube. Microstructurally, the feedwater tube was ferritic/pearlite and fine grain, typical for type SA 210 carbon steel.

GA concluded that the tube sections of both the 800 alloy steam generator tube and ferritic steel feedwater tube are in good condition although the thick magnetite film on the feedwater tube suggests that it may be necessary to chemically clean the tubes in the future. It was recommended that an effort be made to reduce the copper, oxygen and chloride content in the feedwater to control magnetite growth. Based on these examinations, the licensee concluded that the actual cause of the leak could not be determined and it appeared to be random in nature.

The staff didn't concur that the tube leak was a random occurrence and recommended that in the event further leaks occur some form of NDE be conducted to assess the extent of damage.

## Discussion

In response to the staff's concern regarding the ability to conduct non-destructive examinations of the steam generator tubes in the event of future tube leakage, the licensee submitted the referenced report and accompanying documentation with their January 20th, 1984 letter. In the reference report, Public Service Company of Colorado and GA Technologies evaluated the two (2) tube leaks in the Fort St. Vrain steam generators. The first leak occurred in November 1977 and the second in December 1982. Both leaks occurred near the bottom of superheater 2; 1977 in loop 1 and 1982 in loop 2. Both leaks were found at or near a floating tube support plate at about the same elevation.

In order to determine the cause of the tube leaks, the licensee considered all potential factors including residual stresses in the tube bends, weld joint defects, vibration stresses causing fatigue, water chemistry, corrosion, wear, cold springing, low cycle fatigue, crack propagation and loss of tube sleeves and wedges.

The licensee concluded that there is no evidence that any of the above factors were responsible for tube degradation and leakage. However, the coincidental locations of the two tube leaks at the support plate

location raise the remote possibility that the sleeve/wedge assemblies at these support locations were missing or became loose whereby vibrations due to tranverse flow across the tube bundle could have caused the tube leaks. Until additional leaks occur at similar locations, the described degradation mechanism cannot be verified and the tube leakage cause can therefore be considered unknown.

The licensee concludes that the ability to perform quantitative NDE on the steam generator tubes would be desirable in order to determine whether degradation occurred in the steam generator tubes. However, Fort St. Vrain steam generator tubing is generally inaccessible for tubing inspection due to lack of physical access to the tubing area and unit configuration. There currently is no method available for inspecting steam generator tubes without removing steam generator modules from the prestressed concrete reactor vessel (PCRVR). The tubes are not accessible from the primary side due to the shroud design which surrounds the tubes and cannot be inspected internally using current technology, because of the tube design (helical tube bundles, varying tube I.D. and 90° turns at the tube to header or subheader junctions). Although the PCRVR was designed with provisions for removal and replacement of steam generator modules, it would be a difficult, costly and time consuming task. Furthermore, the method has not been demonstrated



nor is equipment available to do the job. Therefore, non-destructive examination of the steam generator tubes is considered impractical at this time and cannot be used to verify tube integrity.

The only areas where NDE is practical are not wholly representative of the tube leak area. These areas are external to the PCRV. The subheader tubes in these areas are made accessible for NDE in the process of capping the subheaders containing the leaking tube(s).

Immediately following each future tube leak, the licensee proposes to perform a metallographic examination of specimens taken from the accessible subheader tubes that are connected to the inaccessible tubes which contain the leak. The results of these examinations will be compared to those obtained from the specimens taken from the tubes that are connected to the previous tube leaks. The licensee will also evaluate the size and elevation of all future tube leaks to determine if additional evidence or circumstances can help to identify a cause or trend in the degradation of the tubes of the Fort St. Vrain steam generators.

### Conclusions

The staff find that since both tube leaks were similar in magnitude and located at or near a tube support plate, they may not be random in nature. However, the staff agrees that there is no practical NDE method for

examining the steam generator tubes due to inaccessibility, helical configuration, 90° turns and varying tube inside diameters along the tube length. In view of the fact that the calculated flaw size in the leaking tube was only 0.003 inch diameter in a 0.205 inch thick tube wall, we do not believe that tube rupture was imminent or that structural integrity of the tubes has been impaired. Furthermore, since through-wall tube penetration results in secondary (water) to primary (helium) inleakage, we do not believe that there is any risk to the health and safety of the public where tube leakage occurs; there is however, an economic penalty for the licensee. Based on these conclusions, the staff does not recommend imposing or implementing scheduled or unscheduled inservice inspection of the steam generator tubes but recommends continuation of primary side moisture monitoring and radiation monitoring of the secondary coolant system as a means of initiating corrective action in the event of steam generator tube leakage. In addition, future post-leakage evaluations proposed by the licensee are acceptable.

ADMINISTRATIVE CONTROLS

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PROCEDURES AND PROGRAMS (Continued)c. Secondary Water Chemistry (PWRs only)

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical variables and control points for these variables,
2. Identification of the procedures used to measure the values of the critical variables,
3. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,
4. Procedures for the recording and management of data,
5. Procedures defining corrective actions for all off-control point chemistry conditions, and
6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.