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

SAFETY EVALUATION BY THE NUCLEAR REGULATORY COMMISSION

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE DPR-34

PUBLIC SERVICE COMPANY OF COLORADO

FORT ST. VRAIN NUCLEAR GENERATING STATION

DOCKET 50-267

INTRODUCTION AND BACKGROUND

Public Service Company of Colorado (PSC - the licensee) provided the results of their evaluation of a Fort St. Vrain (FSV) steam generator tube leak which occurred in December 1982 by letter dated January 20, 1984 (P-84028). We reviewed this report and provided our agreement with the PSC actions by letter and safety evaluation dated June 22, 1984. We did, however, request that the described examination program be incorporated into the Technical Specifications (TS) along with a secondary coolant chemistry program similar to that for light water reactors. PSC responded to our request by application dated August 23, 1984. The initial review of this application identified a typographical error which could have had a significant effect on the interpretation of the required examinations. During discussions with the licensee on the application, it was further decided to expand the basis of the requirement to provide a better explanation. PSC corrected the typographical error and provided an expanded basis by letter dated October 12, 1984.

Subsequent discussions on secondary coolant chemistry requirements resulted in an agreement to delay further action pending receipt of the finalized guidelines presently being developed.

EVALUATION

Following the discovery of a steam generator tube leak on December 8, 1982, PSC investigated and evaluated the problem and then isolated the leaking tube(s) by removing a short section of both the inlet (feedwater) and outlet (steam) subheader tubes and capping both ends. A report discussing the 1982 tube leak and the previous (November 1977) leak was submitted by PSC letter dated January 20, 1984. We reviewed the report and provided our evaluation in a letter dated June 22, 1984, which also requested incorporating the program into the TS. PSC's August 23, 1984 application responded to that request.

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We have reviewed the application and find that it is responsive to our request in that it incorporates the examination and evaluation requirements we have previously approved in our June 22, 1984 safety evaluation. There were, however, two problems with the change as proposed; the first being a typographical error, the second being an incomplete basis for the requirement. These problems were discussed and it was agreed that the typographical error would be corrected and the basis would be expanded to better explain the reasons for the requirements. PSC, by letter dated October 12, 1984, provided a revised submittal to incorporate resolution of the problems.

Therefore, since the requirements added by this application fulfill our request to include previously approved examinations and evaluations and will provide information on steam generator tube integrity, we find it to be acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment changes an inspection or a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 9, 1984

The following NRC personnel have contributed to this Safety Evaluation:
Philip C. Wagner

Attachment: June 22, 1984 Safety Evaluation

Enclosure

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 (PCRV). 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 PCRV 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 impared. 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.

Dated: June 22, 1984

The following NRC personnel contributed to this Safety Evaluation:
L. Frank, NRR
P. Wagner, Region IV