PDR

MONTHLY HIGHLIGHTS

for

Program: Stress Corrosion Cracking of PWR Steam Generator Tubing*

Principal Investigator: D. van Rooyen

Reactor Safety Programs
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

*Work carried out under the auspices of the U. S. Nuclear Regul..tory Commisison

MONTHLY HIGHLIGHTS

for

October 1982

RES Program - Stress Corrosion Cracking of PWR Steam Generator Tubing

D. van Rooyen

Objective:

BNL is developing a model for predicting service performance of Inconel 600 steam generator tubing. The model is based on experimental determination of relationships between factors influencing the stress corrosion cracking (SCC) of tubing in high temperature, deaerated aqueous media.

Progress During This Month:

The long-term exposures at 315°C and 290°C of multiple samples of U-bends of 0.01, 0.02 and 0.03% C materials in pure water continued. At the most recent inspection, a possible small crack was seen but verification by metallogaphy is needed to confirm that it is an intergranular stress corrosion crack. This test was at 315°C, and the result of microscopic examination will be reported next month. These tests are intended to provide the statistical data needed to complete the Arrhenius relationship over the temperature range of 290°C - 365°C. The initiation SCC at these lower temperatures is quite slow, and the first cracks are only expected after two years of exposure. In the case of the 0.01 and 0.02% C alloys, the exposure period for some of our eight surviving (old) samples at 290°C is in the range where SCC may occur very soon, as is the new group of 30 or more at 315°C.

Replicate tests in the constant extension rate test (CERT) apparatus were earlier completed for the temperature relationship of crack growth velocities of as-received Inconel 600 tube specimens in pure water. More points have now been obtained in water with single or paired additions of lithium hydroxide, hydrogen and boric acid, present, as well as simulated primary water at 365°C and 335°C. Arrhenius plots agree with pure water data. H_2 is an adverse addition to pure water, boric acid seems neutral, higher pH is a retarding factor, so that primary water is a cross between being more aggressive due to H_2 and less so due to the LiOH (higher pH). Specimens at two levels of cold work have been made for testing, and initial data show that even 5% cold reduction (in wall thickness) has a pronounced adverse effect.

Replicate tests under constant load continue to confirm the established slopes of log-log plots of failure times vs. applied stress for Inconel in the as-received condition and one level of cold work. Data at two stress levels in simulated primary water agree extremely well with pure water curves, and $b=-4$ in the equation $T_F = k\ (\sigma)^b$, and we are now ready to examine the effect of greater amounts of cold work.

Detailed updates of all the data were made for several purposes, including the Annual Water Reactor Safety Research meeting this year, a paper at next year's annual meeting of NACE, the annual RES summary of accomplishments and a full report of data so far obtained. These will be submitted as needed in the coming weeks.

November 19, 1982

DISTRIBUTION LIST:

G. Arlotto, NRC *

L. Shao, NRC

C. Serpan, NRC

A. Taboada, NRC

W. Y. Kato, BNL

J. Muscara, NRC

U. S. NRC PDR (2 copies)

*Distribution through the office of W. Y. Kato