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Nuclear Energy Research Initiative (NERI)
Quarterly Progress Report

Model Based Transient Control and Component Degradation Monitoring in Generation IV Nuclear Power Plants

DE-FG03-02SF22612/A000

Quarter 1 Report
September – December 2002

Submitted By: The University of Michigan
James Paul Holloway

Collaborating Organizations: Westinghouse Electric Company
and Sandia National Laboratories

Submitted on: February 4, 2003

Preamble:

Our project has had a delay in starting work due to a combination of legal and personnel issues. The non-disclosure agreements required by Westinghouse necessitated considerable legal review; the University of Michigan participants were not able to sign the non-disclosure agreement until the end of October, 2002, and the Sandia National Laboratories was not able to sign until January 2003. In addition, changes in US State Department oversight of visas for foreign students resulted in the graduate students assigned to this project at the University of Michigan to be delayed in their entry into the US. No graduate students were available for this project until January 2003, when one new student became available. An additional graduate student is still required, but none will be available before summer 2003, at the earliest. Both the technical progress and the low expenditures reported below reflect these personnel and legal delays.

The project team held its kickoff program meeting at the Westinghouse Science and Technology Department, George Westinghouse Research and Technology Park in Pittsburgh, PA on November 5, 2002. An overview of the IRIS design was presented. In addition, presentations were made on IRIS instrumentation needs, maintenance optimization and operational control considerations. Unfortunately, some proprietary IRIS information could not be reviewed at that time, as Sandia National Laboratories had not been able to negotiate a satisfactory non-disclosure agreement yet.

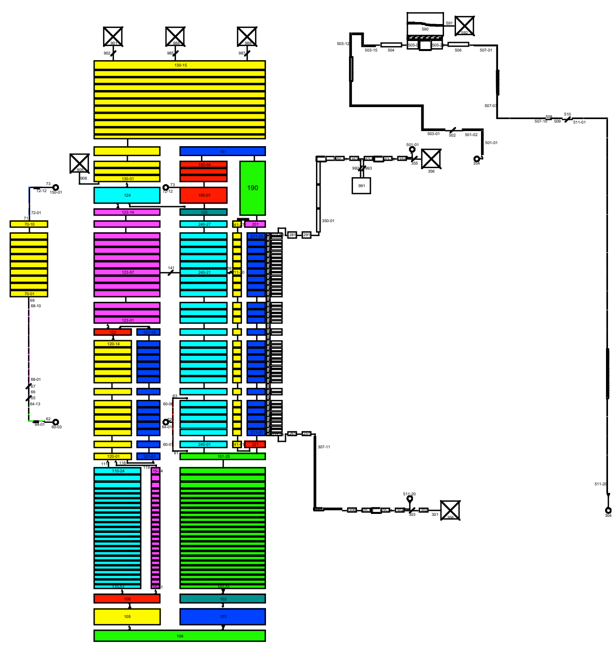
Technical Narrative:

Development of basic nonlinear robust control methodology

Our work in this task to date has been to begin a review of new nonlinear control algorithms based on energy dissipation and passivity bounds. We have also studied the IRIS systems in order to determine where best to begin our control analysis—some consideration has been given to take on a global control algorithm for the entire IRIS plant, but this appears unwieldy and would provide a poor basis for developing new control algorithms. In such an ambitious approach we would spend significant time modeling plant systems, when our energy would be better focused on the development of the control algorithms. The team has therefore decided to focus on control of systems inside the IRIS pressure vessel, including: steam generators, main coolant pumps, pressurizer, and reactor core. Further modeling effort will therefore focus on these systems.

Steam system RELAP model refinement and modification for control simulator

An initial RELAP model has been identified for studying mass and energy flows in the IRIS pressure vessel. This model is based on a Westinghouse model originally designed for safety analysis, and uses a fairly fine nodalization (a sample is shown below). We must still identify and include features necessary for control and degradation monitoring.



Further development of this model for use as a control simulator and as a component degradation simulator is on-going.

Assessment of component reliability databases

A review of failures in existing steam generators has been undertaken. A draft of this review is attached. The importance of crud buildup on the secondary side of existing U-tube steam generators is noted, and is already being considered in existing IRIS monitoring work. In particular the IRIS steam generators may use online eddy current and electro-magnetic acoustic transducer (EMAT) sensors to look for tube thinning, deformation and magnetite buildup on the inside (secondary side in the IRIS design) of the tubes.

The IRIS steam generators provide an important system for demonstrating a component degradation algorithm, and will likely be the focus of our initial monitoring work. The eddy current and EMAT sensors currently planned will only monitor selected tubes within the tube bundle. A monitoring methodology based on comparing plant behavior to a physical model of expected plant behavior, as we will develop in this project, can complement the local monitor of eddy current and EMAT sensors by providing global information about steam generator performance. We expect to consider tube pressure drop and heat transfer degradation as monitoring parameters; these are expected to provide a surrogate for tube deformation and magnetite buildup.

A primary concern is the apparent lack of good statistical degradation data even on existing systems, let alone the proposed IRIS steam generators. The stochastic monitoring methodology that we are developing requires component state transition probabilities. To date we have not identified useful sources of such data. The effort to identify such transition probability data will continue throughout this year. Other approaches to making up for this lack of data include: 1) making estimates of the needed transition probabilities in order to prove the methodology and also to identify key measurements that Generation IV reactor plant development teams should undertake; 2) pursuing a model based degradation monitoring methodology that does not rely on component state transition probabilities, but focuses on component state determination as part of the system state space identification problem. At present we are following all multiple approaches in parallel: 1) continue to seek data, 2) estimate transition probabilities and 3) deterministic methodology.

Steam generator vibration model

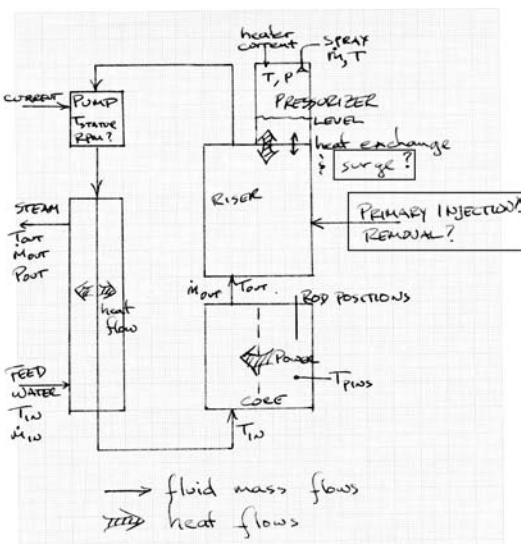
One approach to monitoring for steam generator tube degradation is, as outlined above, to monitor mass and energy flows and compare these to predictions. As another we are studying the use of a tube vibration model. The start of this work has been delayed pending negotiations over non-disclosure agreements. This work will continue at Sandia National Laboratories this year.

Development of model based stochastic degradation monitoring theory

Work has begun on the solution of the stochastic balance equation, similar to a Fokker-Planck or Master equation, that governs the probability density function $p(x,c)$ for plant state, including both the system state x and component state c. A Monte Carlo integration technique is being developed to solve for $p(x,c)$ for multiple diagnostic steps, accounting duly for possible branching out into multiple system trajectories. The solution will be coupled to multiple uses of an adaptive Kalman filter, which may efficiently yield optimal system estimates x and covariance $p(x|y)$ given plant data y, subject to hypothesized component degradations c. Another Monte Carlo algorithm will be implemented to statistically combine $p(x|y)$ with $p(x,c)$ to yield the desired diagnostic information: the probability $p(c|y)$ that the measured plant data are consistent with component degradations c.

Low order and Kalman filtered models of plant dynamics

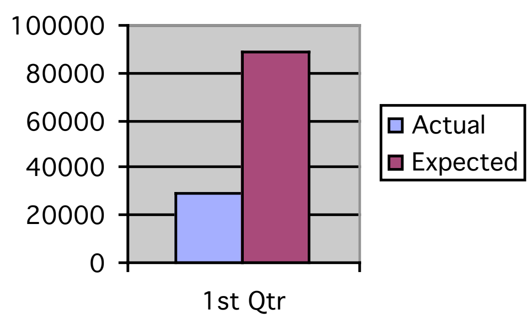
Two approaches are on-going in the development of tractable plant dynamics models. We are considering using a very coarse RELAP model, primarily to estimate how low an order (number of unknowns) is reasonable, by comparing a coarse RELAP model with the more refined RELAP model also under development. A simple set of ODEs representing the mass and energy flows in a simplified model of the plant are also being developed. An initial block diagram for a low order model appears below:



Cost

Performance

U of Michigan – Actual Costs / Anticipated Costs: $6,403 / $52,000
Westinghouse – Actual Costs / Anticipated Costs: $14,209 / $26,250
Sandia -- Actual Costs / Anticipated Costs: $9,000 / $11.000



Issues/Concerns

There are no budgetary concerns at present. As outlined above, the delay in starting work on the project means that we have spent less than anticipated this year. In order to move forward more aggressively the U. of Michigan will increase the appointment fraction for some faculty working on the project during the first two quarters of 2003, in order to make up for the shortage of graduate students originally anticipated for the project.

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Status Summary of NERI Tasks:

Tasks starting year 1	**Planned Completion**	**Actual Completion**	**Percent Complete**
Robust control methodology	September 2003		
RELAP model for control	July 2003		
Assess Reliability Database	October 2003		
Steam generator vibration model	September 2004		
Stochastic degradation methodology	September 2003		
Low order modeling	September 2003		
Main steam system model	March 2003		
Tasks starting year 2	**Planned Completion**	**Actual Completion**	**Percent Complete**
Extend control to MIMO	September 2004		
Develop control simulator	September 2004		
Sensor to state mapping techniques	February 2005		
Develop degradation simulator	January 2005		
Sensor state PDFs	January 2005		
Tasks starting year 3	**Planned Completion**	**Actual Completion**	**Percent Complete**
Control validation	September 2005		
Sensor placement optimization study	July 2005		
Degradation demo	September 2005		

Modes of Failure and Degradation in PWR Steam Generator Tubes

Sumeet Gopwani

INTRODUCTION/BACKGROUND:

Since nuclear power began to be widely used commercially in the 1960's, the most persistent problem with pressurized water nuclear reactors (PWR) has been in steam generators. In a PWR, heated water is carried out of the reactor core through a primary loop to the steam generator. Steam then leaves the steam generator, into a secondary loop, at a temperature of about 500˚ F, and at a pressure much below that at which it entered (2,250 psi). From the secondary loop, the steam goes through a turbine, where electricity is generated, and then through a condenser which turns it back into water. It will then reenter the steam generator to go back through the secondary loop. U.S. PWR's are generally either two-loop, three-loop, or four-loop units.

Westinghouse steam generators generally follow a U-bend recirculating design, where the primary-side water enters at the bottom of the steam generator. The water then flows through tubes that bend in an inverted "U" approximately in the middle of the steam generator, and return to exit at the bottom. Westinghouse units contain about 3,200 to 5,600 tubes per steam generator. Tube diameters range from 19 to 25 millimeters.

Through the 1970's and early 1980's, widespread tube denting as well as various other sorts of tube wear and pitting due to water chemistry issues were the most prominent problem causing tube plugging and wastage. Currently, however, the most common forms of failure is primary-water stress-corrosion cracking (SCC) and outer-diameter stress corrosion cracking/intergranular attack (ODSCC/IGA), as the graph on the right illustrates. SCC is the cracking of steam generator tubes occurring at the tangent point and apex of U-bend tubes, at the tube sheet roll transition zone (RTZ), and in tube dents. It

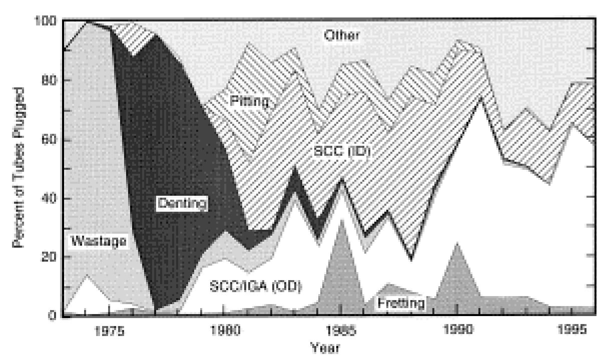

Fig. 1. Causes of steam generator tube plugging. Recently, most common causes of tube failure have become SCC (ID) and SCC/IGA (OD). (Nuclear Engineering and Design, 1999)

occurs when Inconel Alloy 600 tubing is exposed to primary loop water. ODSCC/IGA is caused when tube material is attacked by chemical impurities from the secondary-loop water, and occurs primarily within tube sheet crevices and other areas where impurities concentrate. The remainder of this paper will expand on these modes of steam generator tube failure.

FAILURE MODES:

Over time, a pattern of failures have emerged, and elude to many physical factors which may influence tube integrity. Inconel 600 mill annealed, a thin nickel alloy material, was proven to be susceptible to many forms of cracking, denting, and other types of degradation. Westinghouse has recently evolved to using thermally treated Inconel 690, which has been almost ten times more resistant to cracking. The tube sheet connecting the tubes at each end, which separate the primary-loop water from the secondary-loop water, as well as tube support connections and antivibration bars, also have a tendency to crack over time, all of which can be minimized by higher quality materials and designs which improve steam venting. The tube sheet is the location at which Primary-water SCC most commonly occurs. Cracks are usually axial, but occur circumferentially as well. Circumferential cracking, however, poses much less threat to tube integrity, as long as they are sufficiently contained with in the tube sheet. In U-bend steam generators, the tubes nearest the center of the tube bundle, with the smallest radii, are also susceptible to high stresses and corrosion. This has recently been improved by enlarging the radii, as well as going to Inconel 690 material. Before the material was changed, cracking also occurred commonly in plugs and the material around sleeves, but since then, the problem has subsided.

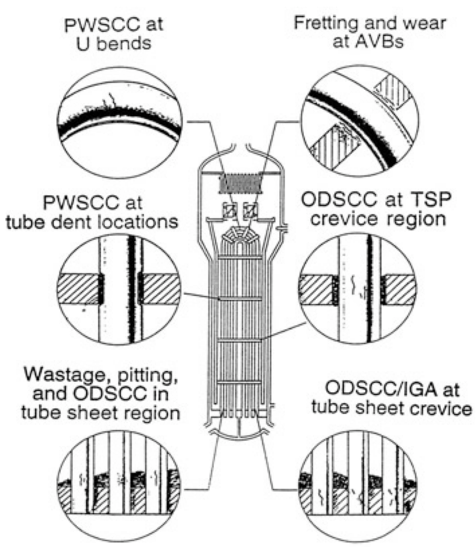

Fig. 2. PWSCC and ODSCC/IGA occur at the apex of U-Bends, tube sheet crevices, and tube dent locations. (US Nuclear Regulatory Commission)

ODSCC/IGA has also been a leading cause of steam generator tube plugging, and has occurred predominantly at the tube support plate (TSP) crevice. Regions above or below the TSP, however, may be sensitive expansion zones. Corrosion products, mainly from the TSP and deposited sludge from the secondary-water system eventually fill the TSP crevice with porous material, restricting secondary-water flow, and increasing concentrations of corrosive species. Most of the porous deposits consist primarily of magnetite. This corrosion commonly causes short axial cracks of varying lengths and depths within the TSP crevice region. Adjacent cracks can often link as material between them corrode, severely compromising tube integrity. This cracking is very common, especially in the hot leg of the U-tube. High concentration levels of dissolved salts in the secondary water can cause high cation/anion ratios, with pH levels exceeding 10, or as low as 2, as opposed to the neutral pH of 5.7 at 300˚ C. These pH extremes can create very aggressive environments with respect to the cracks of the tube materials. ODSCC/IGA at the TSP crevices is most commonly observed in the Westinghouse steam generator designs with carbon steel TSPs and drilled circular holes. This is because the carbon steel plates readily corrode in the secondary-water environment and fill the annular crevices with corrosion product, yet these are still dominant designs in service in the USA. Recent Westinghouse models have used stainless steel and various other treated alloys with improved results in the short run, while improvement in longer service times are yet to be verified. Still, a very shallow crevice is typical in Westinghouse steam generators at the top of the tubes to avoid over-expanding the tubes above the tube sheet, where cracking is possible. Residual stresses introduced by the expansion process at the RTZ appear to contribute to the cracking tendency in this region. Circumferential cracking at the RTZ on both the primary and secondary sides has increased recently, requiring excessive plugging because they can severely compromise tube integrity, and tube reliability may be difficult to determine, and the crack growth rates are difficult to assess.

Generally, the free-span regions of steam generator tubes are considered less susceptible to cracking because of the absence of geometric properties which may contribute to the build up and high concentrations of impurity species, however second-side ODSCC and IGA are not uncommon. Corrosion in these regions may occur for several regions. One possibility is that the geometry of the tube set up creates steam pockets and other sludge build up locations. Otherwise, if chemical species are very aggressive in the secondary water tubes, local concentration may not be necessary for cracking. Frequently, however, these cracks occur simply due to preexisting surface flaws.

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