LWRS Newsletter

Light Water Reactor Sustainability

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Advanced Nuclear Fuel Using Silicon Carbide Cladding to Provide Large Safety Margins

By George Griffith
Advanced LWR Nuclear Fuel Development Pathway Lead

Background

The fuel system, which incorporates the cladding, fuel, and support structure, used in the current generation of light water reactors (LWRs) has proven safe and reliable in all but the most severe events. The modern design of nuclear fuel is reaching maturity and displaying less room for reliable improvement going forward. The top objective of the Advanced LWR Nuclear Fuels Pathway Research and Development pathway within the Light Water Reactor Sustainability (LWRS) Program is to partner with industry and develop fuel with significantly larger safety margins for normal operation and severe accidents. The goal is to develop a ceramic cladding for the fuel that will retain fission products and not react with the cooling water, preventing hydrogen generation at the very high temperatures that occur during severe accidents.

Address Hydrogen Generation due to Zirconium Water Reaction

Advanced nuclear fuel developed by the Advanced LWRS Nuclear Fuel Research and Development pathway must combine features that expand the safety and operating envelope of the nuclear fuel. The advanced design also must maintain or improve the high reliability and predictable operation of the current generation of nuclear fuel. It would be a programmatic failure to develop a fuel that is safe for a specific event, such as the Fukushima Dia-ichi accident, but include features that inhibit routine reliable safe reactor operation. The current generation of fuels is well understood, highly optimized, and a reliable nuclear fuel system. This makes creation of a new nuclear fuel system that is superior in all ways a very difficult task. The difficulty is increased by trying to establish the required understanding of design, performance, and fabrication in a relatively short time. Nuclear fuel vendors anticipate a 10-year cycle of design, testing, and licensing for even small changes in nuclear fuel systems.

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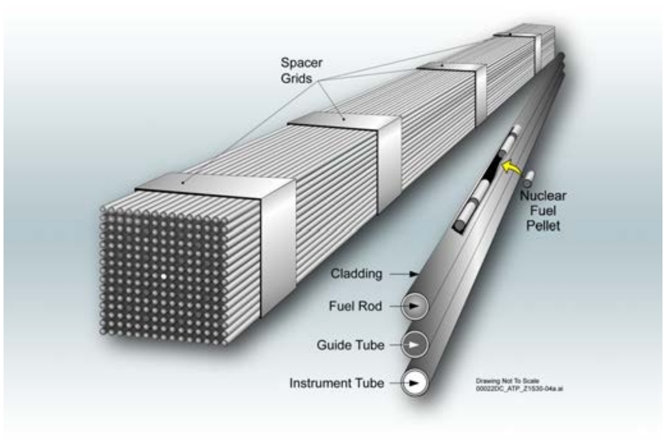
Acknowledging the Fukushima Accident

By Ronaldo Szilard
LWRS Program TIO Director

All of us involved in the activities of the LWRS Program extend our deepest sympathies and most heartfelt prayers to the people of Japan for the tragic events of March 11, 2011. The images of desolation, displacement, and despair resemble a nightmare from which one hopes to awake – only to realize that the injuries and insults to the landscape and its people are real. The extent of damage to the Fukushima Dai-ichi Nuclear Power Plant as a result of the March 11, 2011, earthquake and tsunami continues to challenge our abilities to fully comprehend the impacts of the accident on the local Japanese population and the global community. The disaster further serves to remind us of the serious nature of the LWRS Program's aim to sustain (through science-based research and development) the long-term safe and economic operation of the existing reactor fleet. We will remain diligent in this endeavor, just as the people of Japan remain committed to recovering from this catastrophe.



Nuclear fuel bundle components (a typical pressurized water reactor form).

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Advanced ceramic materials based on silicon carbide (SiC) have been selected as flagship technology to improve nuclear fuel technology. SiC is a ceramic material that has high strength and very low chemical reactivity at high temperatures when compared to zirconium alloys. SiC decomposes at 4900°F compared to a melting point of 3370°F for zirconium. Critically, zirconium also undergoes a water/steam reaction at about 2200°F, leading to failure and production of hydrogen gas. In addition, high-purity SiC conducts heat almost as well as metal, which is used in current nuclear fuel cladding. These properties allow the anticipation of advanced nuclear fuel cladding that can remain intact and safe during and after long periods of time at very high temperatures, such as what is seen in loss-of-cooling events at nuclear reactors or spent fuel pools. The low chemical reactivity of SiC also prevents the energy releasing reaction between very hot steam and zirconium that can occur during severe loss-of-coolant events. This chemical reaction breaks down water molecules into free hydrogen gas, which is a potential explosive when combined with air. The other reaction product is zirconium oxide, which is a brittle ceramic that replaces the original ductile zirconium metal. Both changes (i.e., releasing hydrogen and creating brittle cladding) are detrimental to safe management of a reactor during an accident.

The SiC-based cladding that we are developing is a complex system that uses different materials intended to maximize the operating and accident performance. The initial designs are intended to reliably get test samples into test reactors. This will allow program progress as a final design evolves. SiC fibers, similar in diameter to human hair, are woven into tubular forms that approximate current nuclear fuel cladding. These woven forms are similar to traditional fabric weaves, except SiC fibers are very stiff compared to fabric fibers. The SiC fibers are coated with a very thin layer of soft material, which is an interface layer. Pure carbon is used as the interface layer. The carbon-coated and woven forms are infiltrated with additional solid SiC. The filling layer, or matrix, of SiC locks the fibers in place, creating the very ridged and strong structure expected from ceramic, even at high temperatures. The arrangement of fibers, interface layer, and matrix allows a small amount of stretching not typical of solid ceramic materials. Despite the small amount of ductility, the ceramic material can be expected to generate small cracks under stress. Our design uses a thin inner liner of metal to provide a seal that prevents fission gas from escaping. The metal liner also allows reliable sealing of the tube ends using well known welding techniques. The completed structure is a SiC ceramic matrix composite (CMC) with metal liner.

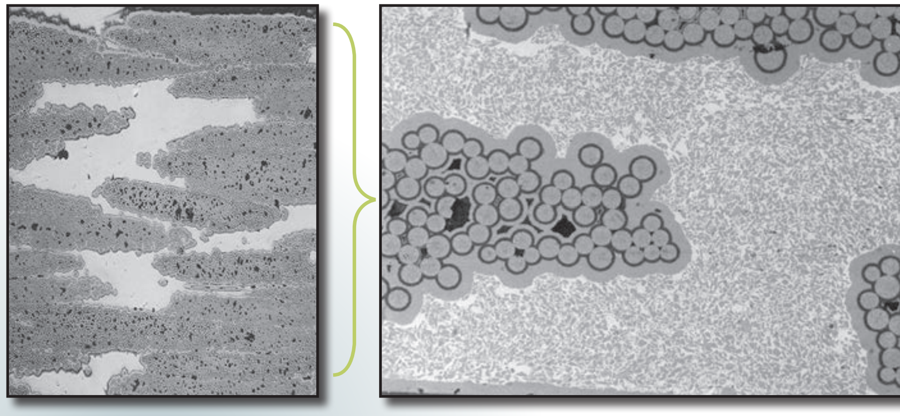
The combination of ceramic fibers buffered from the ceramic matrix allows the cladding to display high strength, stiffness, and toughness not typical of pure ceramics. The ceramic matrix transfers stress to the fibers with some flexibility because of the interface layer, allowing some slip between the matrix and fibers. Failure of the matrix transfers more load to the fibers. Fiber breaks can be accommodated by the load skipping broken fibers across complete fibers. The blunting of cracks and load distribution allows the ceramic to provide a reasonable mechanical performance with the ability to survive at very high temperatures and be chemically inert.

Using a CMC allows many variables to be optimized to produce nuclear fuel cladding with the desired properties. The size, number, and type of weave can control mechanical properties in different directions and change along the length of a sample. Increasing the thickness or type of interface layer affects the fatigue and failure strength of the final composite. Placing additives into the matrix can change the thermal conductivity and enhance the performance at high temperatures. The addition of very small amounts of selected elements will allow the final SiC CMC to produce a self-protecting glass layer at extreme temperatures, extending the performance envelope of the cladding. The large number of variables makes the SiC CMC a deliberately

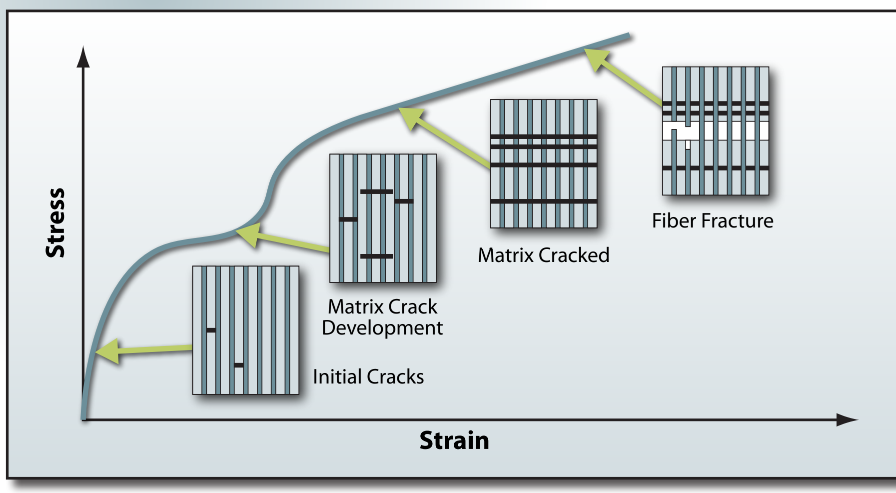
Prototype SiC ceramic matrix composite testing rods.

engineered material, increasing the chances of a successful design.

SiC and SiC CMC systems are established commercial materials used in simple abrasives, chemically resistant coatings, and sophisticated aerospace, rocket, electronic, and turbine engine components. In these applications, the SiC CMC performs in severe environments of temperature, chemistry, pressure, and vibration. We are using this knowledge to advance our understanding of SiC CMC designs and establish technical viability. Commercial nuclear fuel vendors also are cooperating with the LWRS Advanced Nuclear Fuel Research and Development pathway. SiC also has been studied extensively for use as the inner wall in nuclear fusion systems. This research provides information on the stability and properties of SiC when exposed to very high radiation fields. This understanding has helped us to rapidly prototype samples for testing.



Fiber-reinforced material with interface coating around fibers fixed into position with matrix.



Example stress and strain curve for SiC with examples of damage.

Plans to Facilitate Near-Term Commercialization

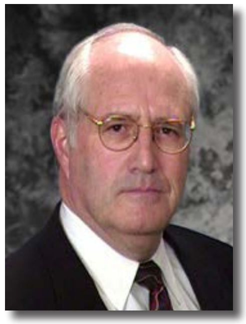
Developing advanced SiC CMC nuclear fuel cladding that is reliable and licensable will require a detailed understanding of the basic materials, fabricated clad tubing, fuel and cladding interactions, and SiC water chemistry in a nuclear field. Research is being conducted at the basic science and applied engineering levels. We are developing integrated computer models that will predict SiC CMC nuclear fuel system performance for the fuel rod, bundle, and whole reactor. This modeling will allow accurate prediction of a specific design performance when complete. We are developing advanced designs that will allow economic and reliable fuel fabrication. New measurement and evaluation tools are being developed to improve understanding of the nuclear fuel cladding.

We are rapidly expanding our nuclear fuel system experimental capability. Multiple mechanical and chemical tests are supporting reactor irradiations. We will use multiple domestic (e.g., Advanced Test Reactor, Massachusetts Institute of Technology Reactor, High-Flux Isotope Reactor) and international (e.g., Halden) test reactors to verify and demonstrate the advanced fuel performance. The reactor testing regime includes steady-state, operating transient, accident, and severe accident scenarios. We plan to predict and test the developed fuel system all the way to failure to ensure the entire performance envelope is understood and predictable. The modeling, understanding, and testing is intended to allow a complete and confidence building licensing basis to be created. We will increasingly depend on nuclear fuel vendors to test, design, and develop the specific commercial reactor application. This is a significant and complex program, which is expected when developing new advanced nuclear technology in a relatively short time. Much of the complexity, expense, and time are created by the need for advanced testing inside test reactors.

The LWRS Advanced Nuclear Fuel Research and Development pathway is actively developing safer and more economic nuclear fuel systems. These advanced designs offer the possibility of much more robust fuel during severe loss-of-coolant events. The integrated plans for modeling, understanding, and testing are intended to provide a complete and trusted technical basis for licensing the advanced fuel. We expect increasing participation of nuclear fuel vendors, eventually leading to fuel designs for deployment in specific commercial reactors.

Integration of Reliability Models with Mechanistic Thermal-Hydraulics Models

Bob Youngblood
Risk-Informed Safety Margin Characterization Pathway Lead



An article in the November 2010 newsletter described R7 as follows:

The centerpiece of the Risk-Informed Safety Margin Characterization (RISMC) pathway is development of the next-generation analysis capability (sometimes called "R7")... This development is aimed at characterizing margin in a "risk-informed" way (that is, based on simulation and analysis of sufficient time histories to sample adequately within the appropriate domains of uncertainty and variability). Correspondingly, it will differ from current capability in several key respects:

a. It is being formulated from the beginning to address uncertainty (e.g., in model parameters) and variability (e.g., in initial conditions and equipment behavior)

b. Numerical methods used in the simulation engine reflect the current state of the art

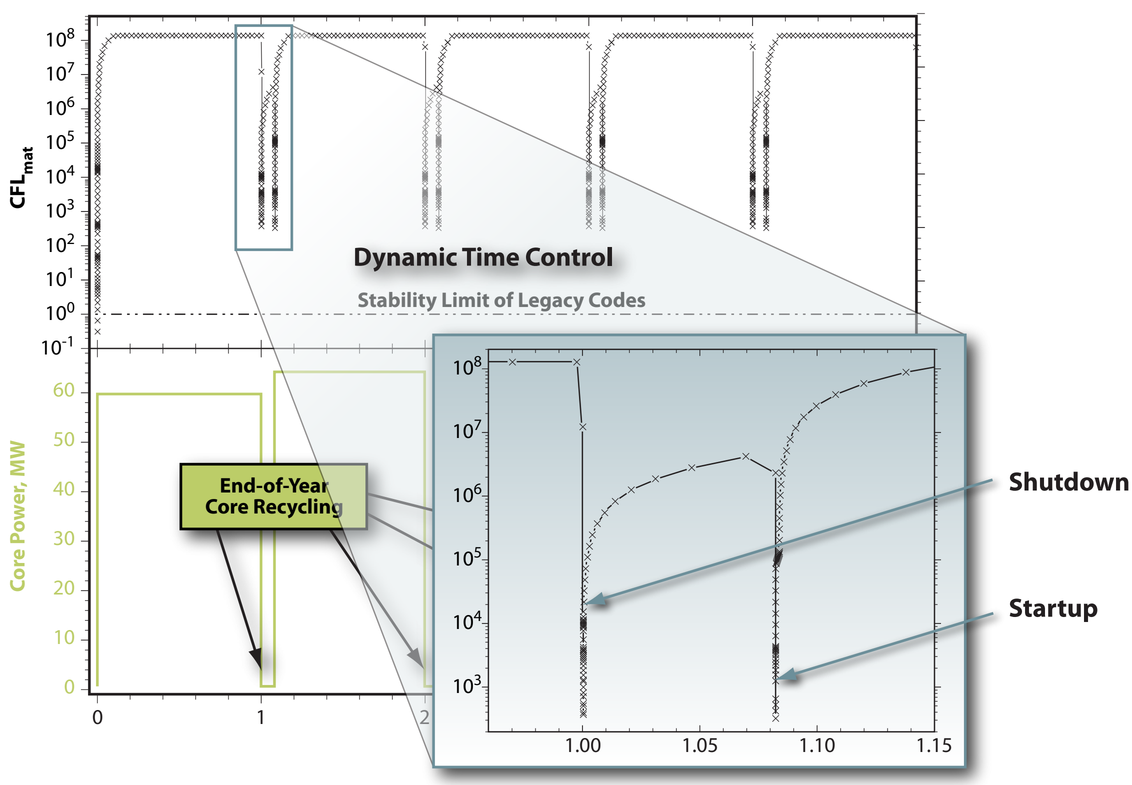
c. The simulation of system, structure, and component (SSC) behavior will be coupled more closely to scenario phenomenology than is practical in today's simulation codes.

The main output of R7 is a characterization of key plant SSC margins in terms of the probabilistic "load" spectrum (showing the range of stresses applied to plant SSCs) and the probabilistic "capacity" spectrum (showing the state of our knowledge of the SSCs' ability to withstand the applied stresses).

In this article, we will describe certain characteristics of R7 that enable this kind of application. In particular, we will describe the integration of reliability considerations and thermal hydraulics modeling mentioned in point c above. This is central to the kind of probabilistic insight that R7 is being designed to provide and especially to the matter of component aging, a point of great interest within the LWRS Program.

Simulation techniques have been applied to both reliability and thermal hydraulics for several generations now. Most of the existing work using simulation to evaluate reliability/availability metrics for complex systems is

Figure 1. An R7 time history spanning multiple years of operation, including refueling outages, showing (a) how time step varies as a function of whether things are changing, and (b) the point that because the solution method is "fully implicit," the fundamental (Courant-Friedrichs-Lewy, or CFL) limit on length of time step is much more liberal in R7 than in legacy codes.



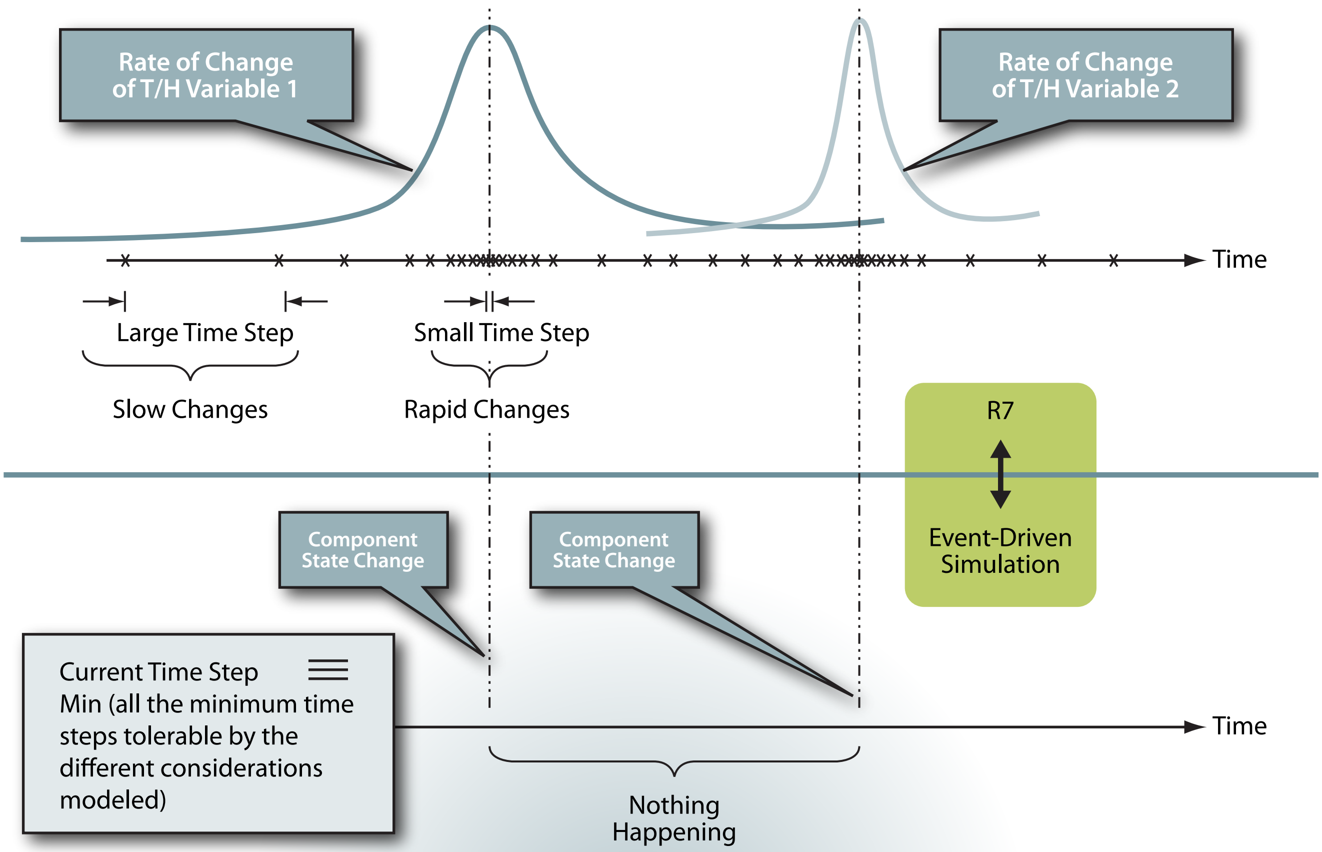


Figure 2. In both R7 and Event-Driven-Simulation, large time steps can be taken when change is slow.

based on simplified models of phenomenology that are implicit in model logic. For example, the logical definition of "success" in terms of combinations of component states is ultimately based on phenomenological considerations, but those considerations are typically (a) simplified and (b) input to the reliability simulation, not done as part of the simulation. In thermal-hydraulics simulations, component behavior is either dictated in the input deck or modeled as part of a "control" system; methods for dealing with aleatory (stochastic) aspects of component behavior in legacy codes are relatively clumsy. In short, integration of the two simulation universes is a challenge. The methodology of legacy codes is not really optimal for uncertainty analysis; it is not easy to blend aleatory models into legacy codes; and, finally, it is difficult to get legacy codes to complete enough runs in a reasonable time to support building up the kinds of distributions of results that are called for in margins characterization.

Certain characteristics of R7 appear to presage a significant step forward in this area. Many simulation codes gain efficiency by relaxing the time step whenever this can be done without sacrificing the accuracy needed in the result; however, because R7 employs high-order-accurate-in-time-and-in-space discretization methods and advanced techniques in linear and nonlinear algebra for solving partial differential equations, the time step allowable in R7 is much greater than in a legacy code. Figure 1 illustrates this point. The horizontal axis is operating time and the vertical axis in the upper plot is given in terms of the "Courant-Friedrichs-Lewy" limit on time step. Underlying this figure is a simulation performed within R7 of a simplified thermal/hydraulics model of a reactor through several years of operation, including refueling shutdowns. The plot in the lower left shows power level as a function of time; the plot in the upper portion of the figure, and the expanded view below and to the right of that plot, show evaluation points. In particular, these plots show that the evaluation points are farther apart when change is slow, but even more importantly, they compare the time step allowable in R7 to the time step allowable in a typical legacy code (shown on the plot as the stability limit of legacy codes). (Note that the scale is logarithmic.)

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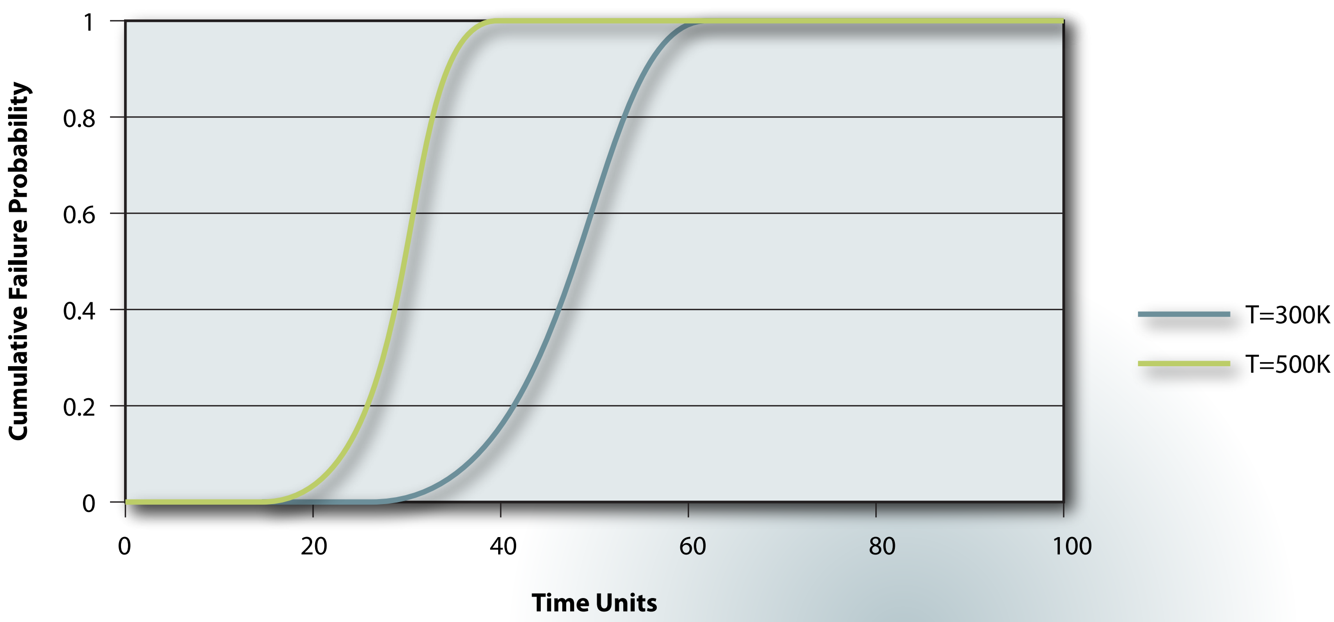

Figure 3. Cumulative Weibull distributions of failure time for two different temperatures (other parameters equal).

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As illustrated in Figure 2 (see previous page), the event-driven approach to reliability simulation is compatible with this kind of time-step control. It turns out that if we know when components are going to fail, it is straightforward to incorporate component failures into the R7 evaluation. In a classical event-driven simulation, we establish each item's failure time at the beginning of each time history and the evaluation of reliability/availability metrics proceeds based on that. However, if we are trying to couple scenario-specific phenomenology to scenario-specific component failure time, as we are in some R7 applications, then we cannot determine failure times at the beginning of each time history.

We can apply the idea of cumulative damage models to solve this problem. In this class of models, components fail when their accumulated damage exceeds a component-specific threshold of damage. From this point of view, classical failure-time distributions for a given component type are understood as aleatory variability in the components' damage thresholds, assuming that damage accumulates at a constant rate. At the beginning of an R7 time history where component behavior is coupled to phenomenology, we cannot sample a failure time distribution to determine validly when a component will fail; but we can sample to determine that component's damage threshold, model the accumulation of damage as a function of environmental stressors (such as temperature) within each time history, and fail the component in the simulation when its damage threshold is reached. In this way, we are able to unify the mechanistic and aleatory aspects of system behavior.

For example, suppose that in a particular component, the accumulation of damage is governed by a phenomenon describable as an Arrhenius process. The Arrhenius model is a widely used idea, according to which an underlying reaction rate is governed by an activation energy EA and the temperature T, thus

$$rate \propto Ae^{-\frac{E_A}{kT}},$$

where

A is a proportionality constant

k is the Boltzmann constant.

For present purposes, we associate damage rate with reaction rate. Within this formulation then, the damage rate at a temperature Tcurrent differs from the damage rate at the nominal temperature Tnom by a factor of

$$e^{\frac{E_A}{k}\left(\frac{1}{T_{nom}}-\frac{1}{T_{current}}\right)}.$$

Based on this idea, Figure 3 compares Weibull distributions of component failure time whose underlying parameters are the same except that they correspond to different temperatures and have different scale parameters given by the Arrhenius model.

Figure 3 illustrates the point that if damage is modeled as an Arrhenius process, a component will fail sooner if the component is operating at a higher temperature. In an event-driven simulation, we would determine each component's failure time by sampling a random number between 0 and 1, and determining the failure time by reading off the time at which the cumulative failure time distribution reaches this sampled value. The horizontal line in Figure 3 corresponds to a sampled random number of about 0.6, which intercepts the T=300K curve at around 50 time units, but intercepts the T=500K curve at around 30 time units.

It turns out that this simple idea applies in a straightforward manner, even if the stressor varies with time (as it generally will), provided that the damage accumulation rate is reasonably constant within a given time step. Moreover, stressors other than temperature (e.g., stress or corrosive environment) also can be modeled in straightforward manner. This creates the potential for addressing an important range of physical influences on component phenomena, including phenomena associated with component aging.

At this writing, simple component models incorporating the cumulative damage idea have been developed and tested within the current version of R7. As is common when a class of models meets with a new application, actual parameter values for particular component types are hard to come by, and this may, for a time, limit the applicability of these models. However, it is reasonable to hope that new data on materials behavior will become available, permitting simulation of passive SSC degradation over multi-year time scales, an accomplishment that will respond directly to a central concern of the LWRS Program.

2011 ANS Annual Meeting "Seizing the Opportunity: Nuclear's Bright Future"

Hollywood, Florida, The Westin Diplomat
June 26 through 30, 2011
http://www.new.ans.org/meetings/c_1

Session title: LWR Sustainability Program Research and Development Overview

As a follow-on to the session held at the ANS 2010 Winter Meeting, this session will provide the latest update on the overview and status of the LWRS Program, which is directed by the Department of Energy and approved by Congress to support the long-term high reliability and excellent performance of the current fleet of 104 nuclear power plants in the United States. Major research and development areas include collaborative research focus areas of Nuclear Materials Aging and Degradation; Advanced Instrumentation, Information, and Control Systems Technologies; Advanced LWR Nuclear Fuel Development; Risk-Informed Safety Margin Characterization; and Economics and Efficiency Improvement. Panelists will address the overall program and major focus areas, including progress-to-date, ongoing activities, and major challenges to operation of our reactor fleet beyond 60 years at very high performance levels. Speakers include

- **Richard Reister** (Department of Energy)
- **Jeremy Busby** (Oak Ridge National Laboratory)
- **Robert Youngblood** (Idaho National Laboratory)
- **Mitchell Meyer** (Idaho National Laboratory)
- **Bruce Hallbert** (Idaho National Laboratory), and
- **Hongbin Zhang** (Idaho National Laboratory).

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Outage Control Center Pilot Project Safety and Efficiency Improvements

By Greg Weatherby
Advanced Instrumentation, Information, and Control Systems Technologies Principal Investigator

Purpose

The Idaho National Laboratory's Human Factors group, working with its industry partner Exelon Nuclear, is deploying new technology to improve outage management at Byron Station. The purpose of the pilot project is to support the commercial nuclear industry by improving both safety and efficiency during nuclear plant outages. Success in this area will enable nuclear utilities to cut costs during outages, shorten outage duration, reduce safety vulnerabilities, and become more cost competitive. These efforts directly support the U.S. Department of Energy's Light Water Reactor Sustainability (LWRS) Program goals to develop technologies and other solutions that can improve reliability, sustain safety, and extend the life of current reactors.

Progress

Exelon's Bryon Station already is an impressive leader in the outage process, averaging 21 days compared to an industry average of 41 days for outages of similar scope. Keith Moser, Exelon's innovation manager and a process improvement team member, sees the future of outage management as a key contributor to being more cost competitive as other non-nuclear energy sources gain prominence. "The future of our industry is dependent on leveraging the advancements in technology with the needs of our future management teams," says Moser. As industry evolves, technology improves, and a new generation of managers and operators emerge, outage teams will expect nuclear plant technology to keep pace with new technology available in the marketplace. Integrated information sharing and prioritization, as well as direct links with suppliers, engineers, vendor technical teams, and management decision makers will increase efficiency. Wireless proximity tracking, workplace three-dimensional modeling, remote dose information monitoring, and heads-up displays are some of the capabilities being looked at by the research team.

Research being conducted by the team leverages the capabilities and experience of some of the best outage managers in the business. Technology is installed and the outage team is trained on its use prior to commencement of the outage. Outage team members use the technology as part of their normal work process and record their experience in a common database for evaluation after the outage. This naturalistic-type approach allows the team to evaluate the effectiveness of the technology upgrade in real-time and to work closely with actual users to develop new and innovative uses to improve processes.

For the March 2011 Byron Station R1 outage, the project team identified a number of early goals it would aim to accomplish. One of the goals was to deploy advanced communication hardware and software designed to improve communication and reduce error. Four SMART Technologies 6052 Smartboards and five Hitachi Starboards were installed and configured to allow the outage control managers and personnel to communicate between remote locations in a collaborative manner and to display critical-path and other important information from separate Exelon databases directly to display mode. INL and Exelon team members will use the information obtained in the March outage to improve outage capabilities for the Byron R2 outage scheduled for fall 2011. Additional technologies and process improvements are scheduled to be deployed through the Department of Energy-sponsored LWRS Program pilot projects and industry partners over the next several years.

Swedish researcher and INL team member, Johanna Oxstrand, readies a touch-capable Smartboard to receive and display critical-path outage information in Exelon's outage control center.

Editor: Teri Ehresman
Writer: LauraLee Gourley

To submit information or suggestions, contact Cathy J. Barnard at **Cathy.Barnard@inl.gov**.