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A Novel Method of Controlling Thermal Hydraulics Codes using RAVEN






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Risk-Informed Safety Margin Characterization Pathway

The Risk-Informed Safety Margin Characterization (RISMC) Pathway develops and delivers approaches to manage safety margins. This important information supports nuclear power plant owner/operator decision-making associated with near and long-term operation. The RISMC approach can optimize plant safety and performance by incorporating physical aging and degradation processes into the safety analysis.

This article describes a novel interaction between probabilistic risk simulation and mechanistic codes for plant-level physics. The new functionality allows the risk simulation module to serve as a "scenario generator" that feeds information to the mechanistic codes. By leveraging progress in the computing science arena, we can model the control logic of a nuclear power plant into a scenario controller for the analysis of risk.

The effort fits with the goals of the RISMC Pathway, which are twofold. The first goal is to develop and demonstrate a risk-assessment method coupled to safety margin quantification. The method can be used by decision-makers as part of their margin management strategies. The second goal is to create an advanced RISMC Toolkit. This RISMC Toolkit would enable a more accurate representation of a nuclear power plant safety margin and its associated influence on operations and economics.

RISMC Toolkit Development

The RISMC Toolkit is a set of software tools used to perform the analysis steps underlying the RISMC method. The tools under development take advantage of advances in computational science and are based on the modern framework, the Multi-Physics Object Oriented Simulation Environment (MOOSE), which was developed at Idaho National Laboratory. These modern tools enable more efficient and accurate modeling than typically afforded by legacy tools.

One component of the RISMC Toolkit is the Risk Analysis Virtual ENvironment (RAVEN), whose development is funded by the Department of Energy Office of Nuclear Energy's Advanced Modeling and Simulation Program. Light water reactors (LWRs) are among its applications. The RAVEN module controls the risk simulation, including generation of accident scenarios. As part of RAVEN's development, we created an enhanced method for controlling physics-based mechanistic codes such as the thermal hydraulics software RELAP-7 (which is under development in the Light Water Reactor Sustainability [LWRS] Program).

A novel interaction between the probabilistic part (i.e., RAVEN) and mechanistic part (i.e., RELAP-7) allows RAVEN

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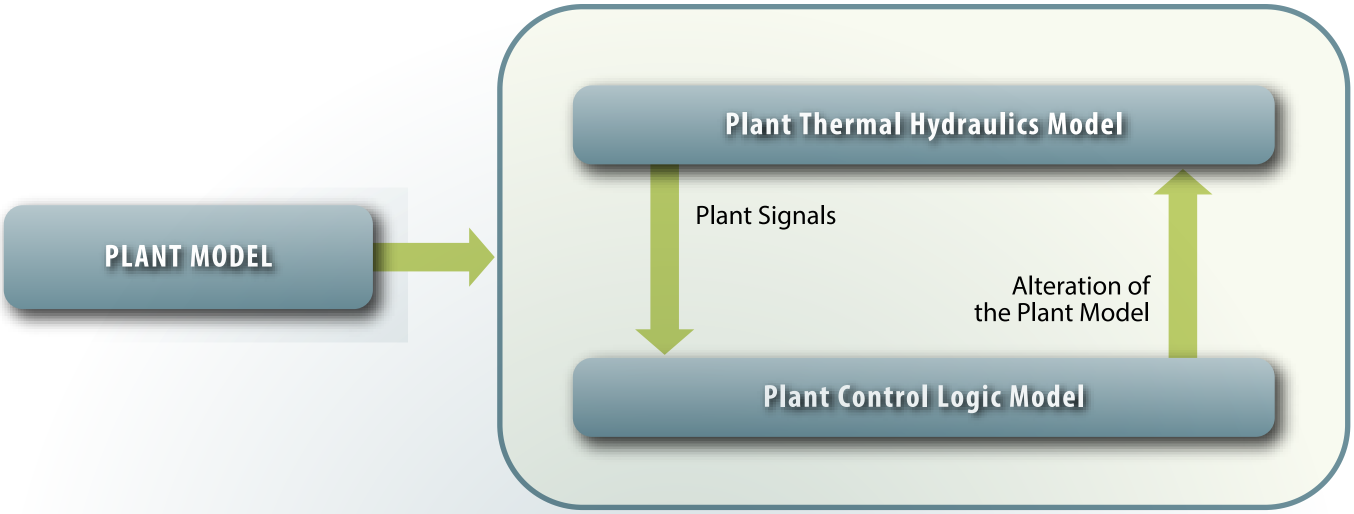

Figure 1. Classic control logic scheme (e.g., RELAP-5).

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to serve as a "scenario generator" within the RISMC approach, where it controls mechanistic codes such as RELAP-7. This feature is critical for simulation-based analysis because the analyst will not know a priori the details of any one particular scenario being simulated. The evolution of the scenario (e.g., timing of events, failure of components, and changes to boundary conditions) is controlled by the RAVEN software.

The RAVEN code is a tool for performing risk analysis and safety margin evaluation for potential off-normal scenarios in nuclear power plants. A key requirement is the ability to control the accident progression simulated by a plant-level code such as RELAP-7. Functionally controlling the simulation is similar to modeling the plant control system and provides a way to enhance realism for safety modeling.

The desire to control deterministic codes drives the need for a very flexible environment (such as the RAVEN control logic system). This system allows for manipulations of the scenario simulation to account for stochastic phenomena, including variations in boundary conditions such as system pressures and temperatures. RAVEN also enables model calibration (for validation purposes) and introduction of other physics models such as proprietary or university-created models.

The Typical Plant Control Logic Model

Historically, the plant control logic model simulation has been kept separate from the thermal hydraulic solver used by the plant level code. The reason they are kept separate is because the mathematical representation of the control logic usually involves discrete functions that are not suitable for the thermal hydraulic code's numerical solver.

In practice, this is implemented via two pieces of dynamic information (shown in Figure 1):

- *Plant Signals*. Readings from the sensor system monitor plant status. Their mathematical representation is the result of manipulation of the thermal hydraulic fields (e.g., temperature or pressure).
- *Alteration of the Plant Model*. Provided by the control logic, these alterations of the plant's numerical thermal hydraulic model aim to introduce changes that are topological (e.g., valve opening/closing) or parametric (e.g., pump speed or boundary condition).

Probabilistic Dynamic Scenario Control

The most relevant difference between probabilistic risk analysis and dynamic probabilistic risk analysis is that component failures may depend not only on a user prescribed (or "average") sequence, but also on the evolution of the plant status. For example, consider the probability of pipe rupture as a function of pressure and temperature changes over time.

Dynamic modeling capability provides a higher fidelity in forecasting risk, but requires a deeper interaction between what is called the scenario generator (i.e., the software that prescribes the plant accident event sequence) and the plant-level simulator. In other words, in dynamic probabilistic risk analysis, component failures may depend on time-dependent plant "signals" and complicated time-dependent failure models and/or recovery possibilities.

As a consequence, in dynamic probabilistic risk analysis, the scenario generator acts more like a scenario controller by providing the dynamic interaction that needs to occur with the plant-level code. A scenario controller has a lot of functionalities in common with what has been described thus far as the "plant control logic model," which we are accomplishing with RAVEN. However, a scenario controller requires some additional capabilities, including information about average pipe pressure and the plant's thermal hydraulic status, which would be provided by RELAP-7.

Software Implementation

The controller built into RAVEN integrates scenario control capabilities with mechanistic codes such as RELAP-7 and other MOOSE-based modules. Figure 2 summarizes this scheme. This interaction happens in a dynamic fashion, meaning that at each time step, MOOSE (1) retrieves the plant status from RELAP-7 and passes it to RAVEN as monitored variables and (2) alters the plant model according to the controlled parameters changed by the RAVEN control logic.

Compared to a classical control logic implementation, which is static, the one available in the RAVEN/RELAP-7/MOOSE environment is fully programmable. The underlying software implementation creates an environment where the variables are available for users to create control logic, ranging from simple to complex.

Further, the control logic is more understandable than traditional approaches, which require input structures that are complicated to both create and understand. RAVEN's Python programming language environment enables users to import and use external functions written in several other languages, allowing an easy way to prototype correlation laws such as heat-transfer attributes or pump-efficiency characteristics.

Implementation Example

To understand the control characteristics built into RAVEN, consider a case simulating a transient that might occur at a pressurized water reactor during a representative station blackout scenario:

- At 100 s, the transient begins (a restart from steady-state conditions previously computed)
- At 101 s, grid power is lost and immediate shutdown of the reactor occurs (scram), followed by
  - Pump coast down
  - Decay heat power
  - Diesel generators fail to operate (AC power lost) and, thus, auxiliary cooling system is inoperable
- At 1675 s, recovery of the AC power and the auxiliary cooling system starts operating
- At 2500 s, transient ends.

For each RELAP-7 component, the controllable and monitored variables are displayed through RAVEN so the user can make modifications through direct interaction or through programmatic rules.

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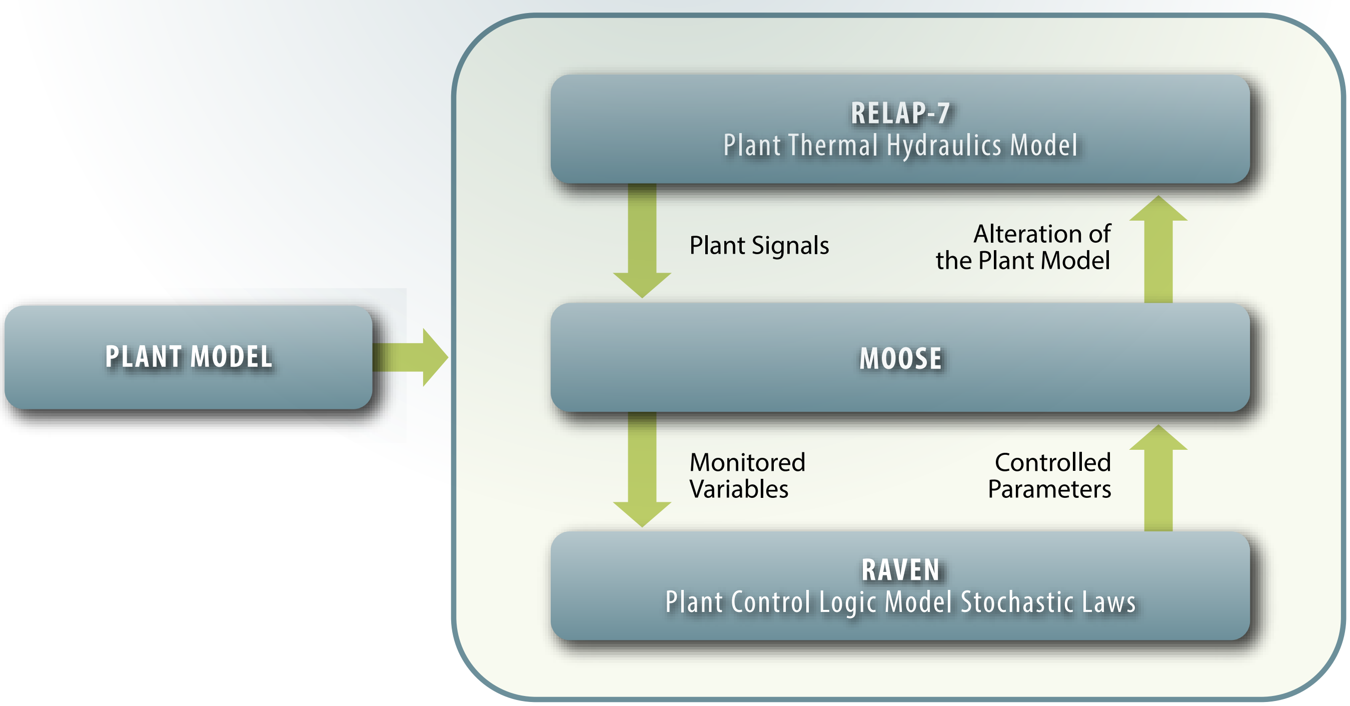
Figure 2. Information exchange flow between RAVEN, RELAP-7, and MOOSE.



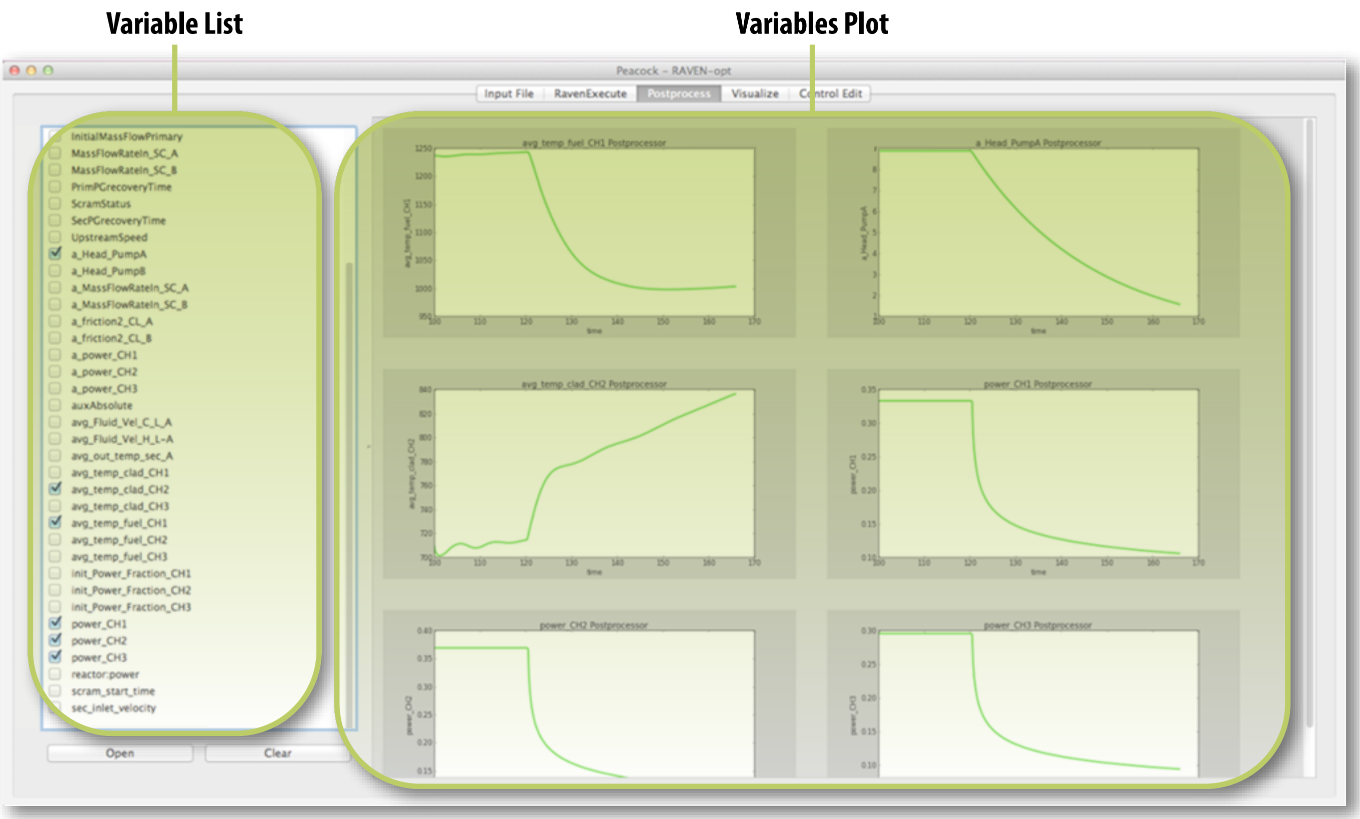


Figure 3. Time evolution of monitored output during transient calculations.

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To perform the station blackout analysis, RAVEN allows the user to emulate different component behaviors as needed. For example, after the loss of power and consequent immediate scram of the reactor, the primary and secondary pumps start to coast down. RELAP-7 then uses these controlled changes (i.e., loss of the pumps) to calculate the reactor power driven by the decay heat.

Results of the control logic application for the transient are shown in Figure 3, where we see the dynamic outputs from various monitored variables.

Implication for the Development of Automated Calibration and Optimization

For code validation/assessment applications, the calibration process will be extremely important. Moreover, it is common that designs need to be optimized to match project requirements. In these respects, the ability to control all parameters characterizing the mathematical model of the plant plays an important role.

For example, the control logic approach built into RAVEN makes it possible to determine the minimum speed of a pump moving the fluid of the secondary side of a heat exchanger so that the primary side fluid is 20°C below the boiling point (or any user-specified margin).

RAVEN is being extended to have an integrated environment, where the controls needed to perform this type of optimization are generated automatically. This optimization module will be the first step toward construction of an environment capable of performing assessment studies using thermal hydraulic models across several experiments in an automated fashion.

Conclusions

The tools and methodologies under development support risk-informed management by providing enhanced methods to control physics-based calculations such as RELAP-7. These approaches will provide important information for owner/operator decision-making associated with the long-term operation of nuclear power plants. Leveraging progress in the computing science arena provides the user with access to powerful methods and tools. RAVEN has been developed to support this research and development goal, with the result of transforming the development of a code to model the control logic of a nuclear power plant into a scenario controller for the analysis of risk.

Nuclear Energy Education Advocate Award

Shannon Bragg-Sitton recently received a Nuclear Energy Education Advocate Award from the Partnership for Science and Technology organization. These awards are presented to individuals or organizations that were central to a noteworthy achievement in energy, nuclear energy, or an environmental field that is of interest to Partnership for Science and Technology members. She was honored, in part, because of her efforts in educating the public and sharing her passion about nuclear energy.

Shannon is the LWRS Program Pathway Lead for the Advanced LWR Nuclear Fuels Pathway and is a key player in transitioning LWRS Program fuels activities to the Fuel Cycle Technologies Program Advanced Fuels Campaign in 2014.

Shannon helped found North American Young Generation in Nuclear (NAYGN) in 1999, which is an organization that provides professional development, public information, knowledge transfer, recruiting, and networking opportunities for the next generation of nuclear leaders. She also served as national president from 2004 to 2005. In 2011, she organized an Idaho Chapter of NAYGN in Idaho Falls. In addition, she is a popular speaker to a wide range of audiences, explaining nuclear and why people should care about it. She especially enjoys speaking with students through programs such as "My Amazing Future," physics teacher workshops, and similar programs. She is one of the featured presenters in a soon to be released video series on "10 Questions about Nuclear Energy" being developed by Idaho National Laboratory.

Shannon Bragg-Sitton recently received a Nuclear Energy Education Advocate Award from the Partnership for Science and Technology organization.

International Collaborative Research and Information Exchange Framework for Irradiated Concrete

Thomas M. Rosseel
Materials Aging and Degradation Pathway

A new international organization, the International Committee on Irradiated Concrete (ICIC), has been formed to provide the framework for exchanging information on a broad set of topics related to the effects of irradiation on concrete. The purpose is to provide a forum for discussing issues that advance the state of knowledge of the effects of irradiation on structural concrete used in nuclear reactor facilities, including storage sites. It is anticipated that the information exchange will leverage capabilities and knowledge, including developing cooperative test programs to improve confidence in the data generated from various concretes irradiated in research reactors.

Understanding the effects of radiation on concrete is important in determining long-term or extended operating performance of concrete structures in existing nuclear power plants. Not surprisingly, this issue is being addressed by research organizations and utilities across the globe (Maruyama et al. 2013, Vodak et al. 2005). Moreover, in the last 2 years, the LWRS Program has been actively working to build international partnerships and collaborations in an effort to better define the issues, develop a sound approach to resolving the major questions, and maximize resources. As part of that effort, an international meeting entitled, "International Irradiated Concrete Information Exchange Framework Meeting," was proposed and organized by Oak Ridge National Laboratory in cooperation with Professor Carmen Andrade, Consejo Superior de Investigaciones Científicas (Spanish National Research Council). Nineteen researchers from five countries attended the meeting, which was held at the Hotel Colon in Barcelona, Spain, on March 12 through 14, 2014. The foundation for this meeting was the understanding that international cooperation will provide the best opportunities to share resources, acquire valuable specimens from decommissioned nuclear power plants, and build a systematic database to provide a framework for decisions concerning extended operation of nuclear power plants in a timely and efficient manner.

Group Photo: Back row: M. Takizawa, J. Zdarek, J. Martin, J. Wall, O. Kontani, C. Castelao, E. Serra, M. Ferreira, J. Sánchez, and T. Eurajoki; Seated: M. Ordóñez, I. Maruyama, K. Field, M. Sircar, T. Rosseel, C. Andrade, and R. Korhonen

The purpose of this meeting was two-fold. The first objective was to develop the framework for exchanging information on a broad set of topics related to the effects of irradiation on concrete used in nuclear power plants by those who are actively pursuing research, were active in the field, or wish to contribute to advancing the current state of knowledge. The second objective was to provide a forum for discussing issues that advance the state of knowledge of the effects of irradiation on structural concrete used in nuclear reactor facilities, including storage sites.

The first portion of the meeting included presentations and discussions on past and current irradiated concrete research and issues related to irradiated concrete. The second portion of the meeting focused on establishing the framework for exchanging information. This included a discussion of the types of information that could be exchanged; the level of release; an organizational framework for cooperation, including resource and data sharing; and development of a charter that is based on the International Group on Radiation Damage Mechanisms in reactor pressure vessels.

The final portion of the meeting focused on making a decision or commitment by attendees to participate in the information exchange. At the conclusion of the meeting, the participants reached a consensus to move forward with the ICIC and to hold a follow-up meeting within 6 months to finalize the charter and elect an executive committee. Moreover, the participants endorsed a plan to determine the frequency and location of future ICIC meetings. It is anticipated that future meetings will be held on a rotating schedule in Europe, the United States, and Japan. The meeting concluded with the election of Dr. Thomas M. Rosseel, Acting Chair, and Dr. Carlos Castelao, Consejo de Seguridad Nuclear, Acting Vice Chair, to lead the interim process. Information on ICIC, the draft charter, and the meeting agenda and presentations can be found on the ICIC web site (http://web.ornl.gov/sci/physical_sciences_directorate/mst/IICIEF/index.shtml).

References:

Maruyama, I., O. Kontani, S. Sawada, O. Sato, G. Igarashi, and M. Takizawa, 2013, "Evaluation of Irradiation Effects on Concrete Structure - Background and Preparation of Neutron Irradiation Test," in *Proceedings of the ASME 2013 Power Conference POWER2013,* Boston, MA, USA.

Vodak, F., K. Trtik, V. Sopko, O. Kapickova, and P. Demo, 2005, "Effect of Gamma Irradiation on Strength of Concrete for Nuclear-safety Structures," *Cement and Concrete Research* 35, 1447-1451.

ICIC, 2014, International Committee on Irradiated Concrete, web site: http://web.ornl.gov/sci/physical_sciences_directorate/mst/IICIEF/index.shtml.

LWRS Program helps Arizona Public Service win Top Industry Practice Award

The Nuclear Energy Institute awarded a Top Industry Practice (TIP) Process Award for Materials, Management Processes, and Support Services to Arizona Public Service Company for leveraging technology to improve outage coordination and performance. The award, presented May 20, 2014, in Scottsdale, Arizona, was for an innovative process improvement employed by Palo Verde Nuclear Generating Station in collaboration with researchers from the LWRS Program to manage and provide enhanced collaboration of information and activities associated with Palo Verde's fall 2013 refueling outage. Idaho National Laboratory (INL) researchers have been working closely with Palo Verde staff for over a year to develop and deploy new technologies to improve refueling outage performance. Palo Verde Nuclear Generating Station team members were recognized at the awards luncheon, along with INL team members, for their role in the technology development and in Palo Verde's successful deployment of the technology, resulting in the TIP award. This is the first time INL staff have been honored in this way and as part of a team recognized with the Nuclear Energy Institute's top distinction.

Arizona Public Service noted that effective management of refueling outages is essential to the long-term commercial viability of nuclear energy facilities. Outage delays incur significant expenses due to the costs of replacement power and additional labor. The Advanced Instrumentation, Information, and Control Systems Technologies Pathway of the LWRS Program, led by Idaho National Laboratory's Bruce Hallbert, has included outage safety and efficiency pilot projects in its research portfolio. These pilot projects demonstrate how the advanced instrumentation, information, and control technologies can improve refueling outage performance. Principal investigators, Shawn St. Germain and Ronald Farris, led research into the processes and methods that the Palo Verde Nuclear Generating station uses to manage outages, emphasizing the human factors involved in communications, coordination, work management, and other essential activities. Working with staff from Arizona Public Service, they developed technologies

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and methods to enhance work management and improve efficiencies in all areas of outage work management.

During Palo Verde's fall refueling outage, a leak was discovered on a reactor vessel bottom-mounted instrumentation penetration. A similar bottom-mounted instrumentation issue had been experienced at South Texas Project. The time from issue identification to repair at South Texas Project was about 72 days. Use of operating experience, vendor support, and enhanced communication, facilitated by this new technology, allowed the Palo Verde Nuclear Generating Station to complete similar repairs in about 32 days. The additional cost avoided by reducing the outage extension from the previous benchmark was estimated at $48M. Several Palo Verde Nuclear Generating Station managers involved in resolution of the bottom-mounted instrumentation issue said that the improved collaboration tools helped them achieve success in issue resolution. The TIP awards recognize achievements in 13 categories, with four reactor vendor awards and nine process awards for innovation to improve safety, efficiency, and nuclear plant performance, as well as an award for vision, leadership, and ingenuity.

Recent LWRS Program Reports

Materials Aging and Degradation

- ***Perspective on Radiation Effects in Concrete for Nuclear Power Plants - Part I: Quantification of Radiation Exposure and Radiation Effects***
  http://www.sciencedirect.com/science/article/pii/S0029549314005603
- ***Perspective on Radiation Effects in Concrete for Nuclear Power Plants - Part II: Perspective from Micromechanical Modeling***
  http://www.sciencedirect.com/science/article/pii/S0029549314005718
- ***Establishment of an International Irradiated Concrete Information Exchange Working Group***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/International_Irradiated_Concrete_Information_Exchange._Report.F.pdf
- ***Comprehensive and Comparative Analysis of Atom Probe, Small-Angle Neutron Scattering, and Other Microstructural Experiments on Available High Fluence Reactor Pressure Vessel Steels***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/Small_Angle_Neutron_Scattering_report-2014.pdf

Risk-Informed Safety Margin Characterization

- ***RELAP-7 Theory Manual***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/RELAP-7_Theory_Manual_INL-EXT-14-31366.pdf
- ***RELAP-7 Numerical Stabilization: Entropy Viscosity Method***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/INL-EXT-14-32352_R0.pdf
- ***Light Water Reactor Sustainability Program Case Study for Enhanced Accident Tolerance Design Changes***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/Case_Study_for_Enhanced_Accident_Tolerance_Design_Changes-INL-EXT-14-32355_R0.pdf

Advanced Instrumentation, Information, and Control Systems Technologies

- ***Light Water Reactor Sustainability Program Operator Performance Metrics For Control Room Modernization: A Practical Guide For Early Design Evaluation***
  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/LWRS_2014_M3_Report_Human_Performance_Measures.pdf

Technical Integration Office

- ***Light Water Reactor Sustainability Program Accomplishments Report, 2013***
  https://lwrs.inl.gov/Technical%20Integration%20Office/LWRS_Program_Accomplishments_Report_2013.pdf

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