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Evaluating the Effects of Irradiation on Concrete at Extended Lifetimes




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Over the last 15 years, more than 70 nuclear power plants in the United States have received approval to extend their license to operate up to 60 years. Extending the operating lifetimes of current nuclear power plants beyond 60 years and, where practical, making further improvements in their productivity is essential to meeting future national energy needs while reducing greenhouse gas emissions. In order to meet these goals, a critical evaluation of the knowledge gaps of materials that comprise the structures and components of a nuclear power plant must be addressed. Although much of the focus has been on the performance and possible degradation mechanisms of metals due to increased exposure time to temperature, stress, coolant, and radiation fields, other materials (such as cables and concrete) also are critical to long-term operation.

For example, large sections of most nuclear power plants have been constructed using concrete. In general, the performance of reinforced concrete structures through the first 40 years of service has been very good. Moreover, reported incidents of degradation often occurred early in the life of the structures and primarily have been attributed to construction/design deficiencies or improper material selection. Although it is expected that the vast majority of these structures will continue to meet their functional and performance requirements during the current and any future licensing periods, there may be isolated examples where structures may not exhibit the desired durability, primarily resulting from environmental effects. To address these potential gaps in the knowledge base, the Materials Aging and Degradation Pathway has established a research plan to investigate the aging and degradation processes associated with the concrete used in nuclear power plants.

As shown in Figure 1, additional information is needed on the effects of irradiation on concrete structures (such as the biological shield) to demonstrate that the structures will continue to meet functional and performance requirements under long-term operation. In an effort to better understand and manage potential degradation, Kontani et al. (2011) reviewed the literature on irradiated concrete. However, much of the historical data of irradiated concrete does not accurately reflect the types of concrete used in nuclear power plants and may not reflect typical irradiation conditions (Fillmore 2004). To address these concerns, a five-pronged approach was formulated in partnership with the Electric Power Research Institute. This

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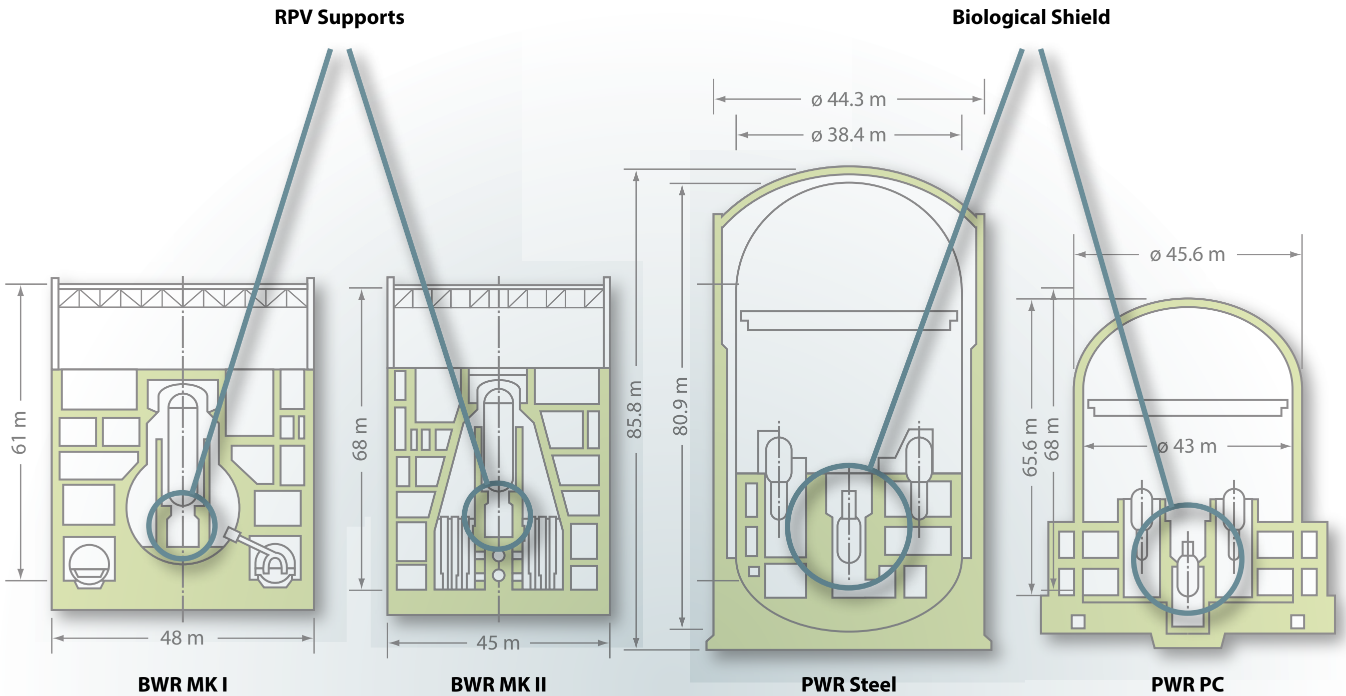

Figure 1. At extended operation greater than 60 years, will gamma and neutron exposure levels be sufficient to impact the concrete shielding effectiveness or load-bearing capacity?

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article will focus on developments within these five prongs, including (1) assessing the radiation environment in the concrete biological shields of nuclear power plants, (2) performing accelerated irradiation studies of prototypical concrete to levels equal to or greater than expected in extended service, (3) harvesting and evaluating irradiated concrete from nuclear power plants, (4) utilizing modern materials characterization techniques to develop a fundamental understanding of the effects of radiation on concrete, and (5) establishing collaborative research efforts with international partners and/or universities to enhance our ability to understand and bound the potential damage.

Assessment of the Radiation Environment in Concrete Biological Shields of Nuclear Power Plants

The first step in understanding the effects of radiation on degradation of concrete in nuclear power plants under extended service conditions is to determine the neutron fluence and gamma-ray dose in the biological shield at 80 years of operation and beyond. These estimates will provide a valuable and conservative tool for utilities and regulators to assess the scope of the issue, as well as provide the boundary conditions for performing irradiation studies on prototypical concrete and for efforts to develop a fundamental understanding of the effects of radiation on concrete.

While there is no routine monitoring of radiation fields in biological shields of nuclear power plants, valuable information can be obtained from the reactor pressure vessel surveillance programs, which are required for every operating nuclear power plant and are especially valuable when coupled with dosimetry measurements outside the reactor pressure vessel (ex-vessel dosimetry). The initial work on how these data can be extrapolated to guide the accelerated irradiation of concrete and perform screening of the nuclear power plants is discussed in the following paragraphs.

Coupled neutron and gamma transport calculations for one selected, three-loop and one, two-loop pressurized water reactor were conducted recently at Oak Ridge National Laboratory. An example of the neutron and gamma-ray flux variation through the pressure vessel and biological shield is shown in Figure 2.

Observations and conclusions from these radiation transport simulations of neutron and gamma fields are as follows:

- Fast neutron (E > 1 MeV and E > 0.1 MeV) fluxes at the pressure vessel outer radius are 20 to 30% higher than the maximum fluxes in concrete. These fluxes, determined at maximum at the pressure vessel wall, are typically provided in reactor pressure vessel surveillance reports. These values could be used as conservative estimates for fluxes in the concrete shield or for screening purposes.
- The total neutron and gamma flux at the pressure vessel wall agree within 10% of the maximum values in concrete. Heating rates calculated at the pressure vessel maximum flux location also are 10 to 20% higher than the highest rates in concrete.
- It is not practically possible to renormalize the results from the fast neutron fluence (E > 1 MeV) to total neutron fluence or vice versa without knowing the details of the irradiation experiment.
- Although two-loop and three-loop pressurized water reactors have quite similar neutron and gamma-ray spectra in the cavity region, the analyzed two-loop plant has 2 to 4 times higher neutron and gamma fluxes and heating than the considered three-loop plant. The four-loop plants are expected to have lower fluxes outside the pressure vessel; however, this needs to be confirmed.

Table 1 lists the number of years of operation in which the accumulated neutron fluence in the concrete shield (at the most exposed location and assuming a 92% capacity factor) will reach 1, 5, or 10 x 10^{19} n/cm,$^{-2}$ for neutron fluence cutoff energies of 0, 0.1, and 1.0 MeV. Both two and three-loop plants reach the total (E > 0 MeV) neutron fluence of 1 x 10^{19} n/cm^{-2} early in their lives, at 5 and 11 years, respectively. For the fast fluence (E > 0.1 MeV) both plants exceed 1 x 10^{19} n/cm^{-2} value during normal lifetime; however, it takes the two-loop plant 123 years to reach the fast fluence (E > 1 MeV) 1 x 10^{19} n/cm^{-2}, which is longer than the currently anticipated extended operation and even much longer for the three-loop plant. To reach a gamma-ray dose of 1 x 10^8 Gy, it takes 70 years for a two-loop plant and 172 years for a three-loop plant.

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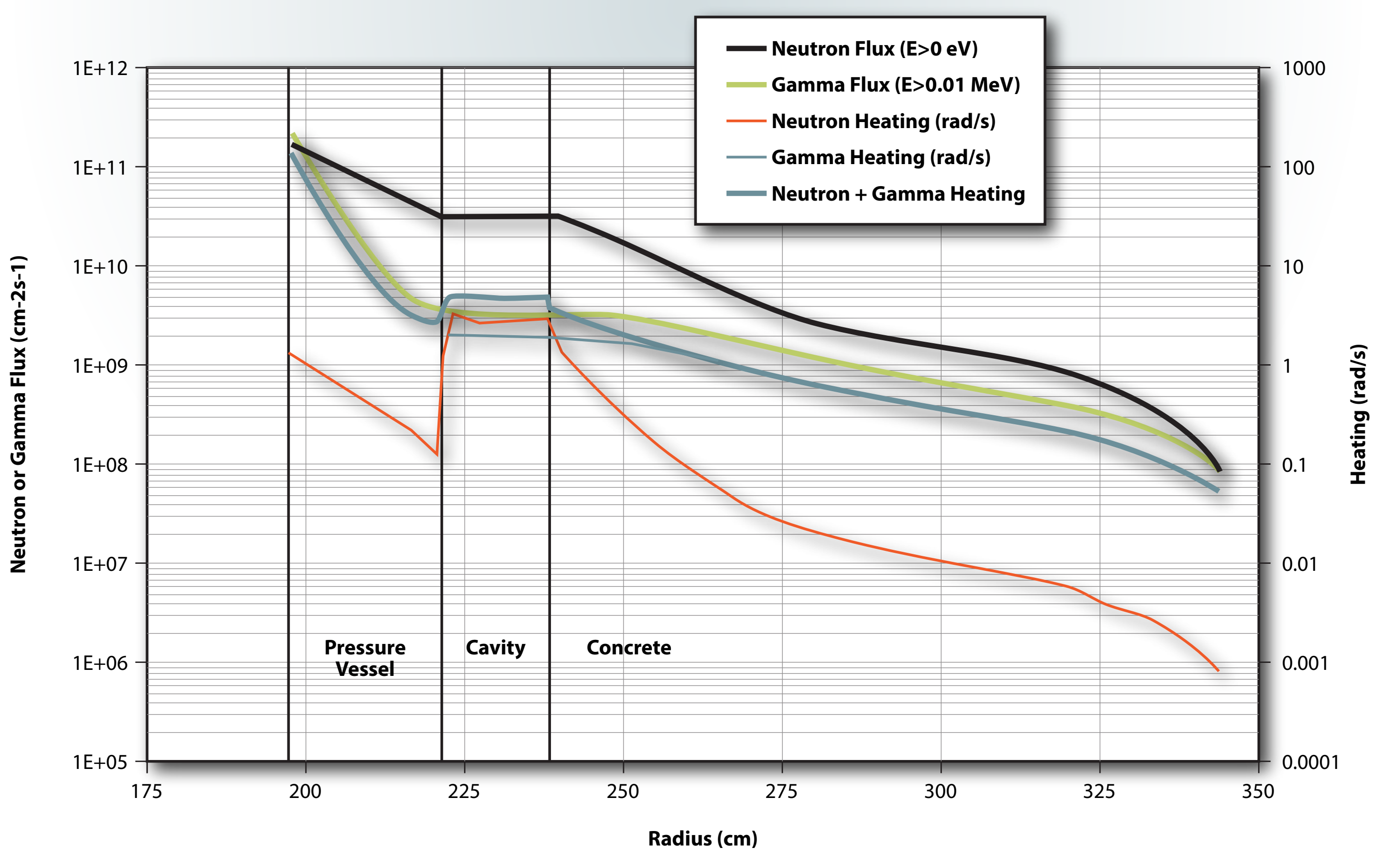
Figure 2. Radial variation of neutron and gamma-ray flux and heating rates for a three-loop pressurized water reactor.



Calendar Years of Operation				
Fluence (1×10^{19} n/cm^{-2})	Pressurized Water Reactor Type	Neutron Fluence Cutoff (MeV)		
		E > 0	E > 0.1	E > 1.0
1	Two-loop	5	14	123
	Three-loop	11	30	
5	Two-loop	27	71	
	Three-loop	56	152	
10	Two-loop	55	143	
	Three-loop	111		

Table 1. Years of operation (at 92% capacity factor) to reach neutron fluence of 1, 5, or 10 x 10^{19} n/cm^{-2}.

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Clearly, if a threshold fluence above where the deterioration of concrete occurs can be specified in terms of fast neutron fluence (E > 1 MeV), then data from the reactor pressure vessel surveillance programs should be sufficient to assess the conditions of concrete in nuclear power plants. However, if gamma rays, synergistic neutron and gamma ray effects, or neutrons at energies below 1 MeV also are important, additional plant-specific transport calculations will likely be necessary because the reactor pressure vessel surveillance programs do not contain such information.

These preliminary results demonstrate the importance of specifying the fluence cutoff energy when determining the threshold fluence above where degradation is expected, performing accelerated irradiation experiments on concrete, or assessing operating nuclear power plants. For a selected neutron fluence cutoff energy, there is a linear relationship between the magnitude of the threshold fluence and the years of operation to reach that threshold.

Accelerated Irradiation of Prototypical Concrete

The data on degradation of concrete due to long-term irradiation are quite limited. For example, Fillmore (2004) concluded, based mostly on the somewhat dated Hilsdorf et al. (1978) compilation, that no effects of radiation on strength were found for neutron fluence less than 10^{19} n/cm^{-2} or gamma ray doses of 10^8 Gy for periods less than 50 years. Reductions in compressive and tensile strength and a marked increase in volume are reported for neutron fluences greater than about 10^{20} n/cm^{-2} or gamma-ray doses of 2×10^8 Gy. However, data at intermediate fluences and doses are not consistent. For many older experiments, important information (such as the cutoff energy of the neutron fluence reported, the composition of concrete, irradiation temperature, and gamma ray dose) is often limited or missing. Consequently, the applicability to nuclear power plant concrete is uncertain. Recent experiments by Fujiwara et al. (2009) appear to support Hilsdorf et al. (1978) data for neutron irradiations, while results presented by Vodak et al. (2005) appear to show degradation of concrete at 3 to 5×10^5 Gy gamma ray doses. Clearly, more data are needed; however, based on the conflicting results from the limited data and assessment of the radiation environment in the concrete biological shield described in the previous section, planning of experiments with accelerated irradiation of concrete should consider the following critical issues:

- It is desirable to obtain full neutron and gamma spectra in the samples for irradiation experiments because the relevant irradiation parameters for concrete have not been established.
- Irradiation experiments should be designed so that the neutron and gamma spectra in the concrete samples will be similar to those observed in nuclear power plants biological shields.
- Accelerated irradiation experiments will require considerably higher neutron and gamma fluxes than those observed in the shields on nuclear power plants. The acceleration factor and, therefore, fluxes are anticipated to be higher by a factor of 50 to 100. Analysis of the temperature of the samples will be necessary and additional cooling of concrete samples may be required. Moreover, the effect of dose rate has not been evaluated.

High-flux neutron irradiation facilities available at research reactors provide an opportunity for accelerated irradiation of concrete. These facilities potentially can meet the entire criteria for accelerated testing and, in fact, groups from Japan and Finland have initiated experiments at the Institute for Energy Technology's JEEP-2 reactor facility at Kjeller, Norway, to address the paucity of reliable data.

Other options include the LVR-15 reactor at the Research Centre Rez, Czech Republic, the Advanced Test Reactor at the Idaho National Laboratory, or the High-Flux Isotope Reactor at Oak Ridge National Laboratory. Moreover, irradiations at research reactors could provide similar fluence/heating rate versus depth curves as observed in biological shields of nuclear power plants, which would help develop testing techniques for harvested cores as discussed in the next section.

Harvesting and Characterization of Irradiated Concrete from Nuclear Power Plants

During the last decade, several nuclear power plants in the United States and other countries have been decommissioned, are in the process of being decommissioned, or decommissioning will commence soon. Examples include Zion Nuclear Power Station Units 1 and 2, Milestone 1, Indian Point Unit 1, Crystal River 3, Zorita (Spain), nuclear power plant Krummel (Germany), and Barseback (Sweden). Harvesting of concrete cores from decommissioned nuclear power plants will provide an opportunity to generate data from concrete that has experienced typical radiation fields, while also providing guidance to accelerated irradiation studies. The coupling of accelerated or laboratory-irradiated concrete with harvested nuclear power plant cores is expected to facilitate the effort to develop an understanding of the damage mechanisms in irradiated concrete, including understanding the potential effects of accelerated testing.

In support of extended service of the U.S. nuclear reactor fleet, Oak Ridge National Laboratory (through the Light Water Reactor Sustainability [LWRS] Program) is coordinating and contracting with Zion Solutions, LLC (a subsidiary of Energy Solutions) the selective procurement of materials, structures, and components, including concrete from the decommissioned Zion reactors (Rosseel et al. 2012). Physical acquisition of concrete cores will need to be supplemented by extensive investigations into the operating history of the nuclear power plant, as well as the concrete composition and performance history. Information such as the material test reports for cement, admixtures, and aggregate; concrete mix design; and aggregate characterization, including type, petrographic analyses, and gradation will be collected. Other critical information would include concrete property test results (e.g., reference 28-day compressive strength and modulus of elasticity), results from any concrete strength testing over the life of the plant, and the American Society of Mechanical Engineering inspection reports. It is anticipated that data from the concrete from the Zion nuclear power plant will be integrated with data from other decommissioned reactors and possibly with data from accelerated irradiation of well-characterized nuclear power plant-like concrete.

As noted previously, the neutron and gamma flux, as well as heating effects, decrease as a function of distance from the core, with a one-order of magnitude drop at about 25 cm from the inner surface of the biological shield for the neutron flux. This gradient effect is highly beneficial in assessing the effects on concrete because different locations along a core, which can be on the order of a meter or more, provide specimens at different neutron fluences and gamma doses after serial sectioning. Therefore, from a single core, trend curves of mechanical properties versus fluence or dose can be generated. These results can be correlated with accelerated testing programs to extrapolate the effect of 80 years of operation on the mechanical performance of concrete.

Developing a Fundamental Understanding of the Effects of Irradiation on Concrete

Concrete is a complex, multi-component, composite material that must be understood across multiple length scales. Additionally, no formally accepted testing protocol (such as an ASTM standard) currently exists for irradiated concrete. Provided that representative samples of sufficient size having uniform exposure to neutron fluence or gamma dose are available, traditional test methods will be used to determine the mechanical properties of interest (such as the concrete tensile and compressive strengths, modulus of elasticity, dimensional change, and weight or mass change). However, because neutron and gamma flux, as well as heating, may exhibit steep gradient through the concrete samples, small-scale methods (such as nano-indentation) may need to be employed to assess the mechanical properties.

Chemical composition, porosity, and density also will need to be determined. These factors can be related to the mechanical performance tests to determine which variables most significantly affect radiation-induced degradation. Traditional routes (such as petrography) will be explored as a preliminary screening tool. If small-scale testing techniques are deployed for mechanical testing, traditional petrography and characterization techniques might be insufficient. Thus, advanced analytical microscopy techniques currently are being explored to assist in characterizing irradiated concrete.

One such technique, which has been used to great benefit in other irradiated material systems, is focused ion beam (FIB) coupled with scanning electron microscopy (SEM). FIB-SEM allows for site-specific investigations at high spatial resolution. Moreover, FIB-SEM can generate cross sections through interfaces and transmission electron microscopy samples and can be utilized for serial sectioning to generate three-dimensional tomography images. These techniques may assist in investigating the radiation effects (such as aggregate expansion and cementitous contraction or alkali-silica reaction) that could lead to cracking at the interface between the aggregate and cement paste in irradiated concrete.

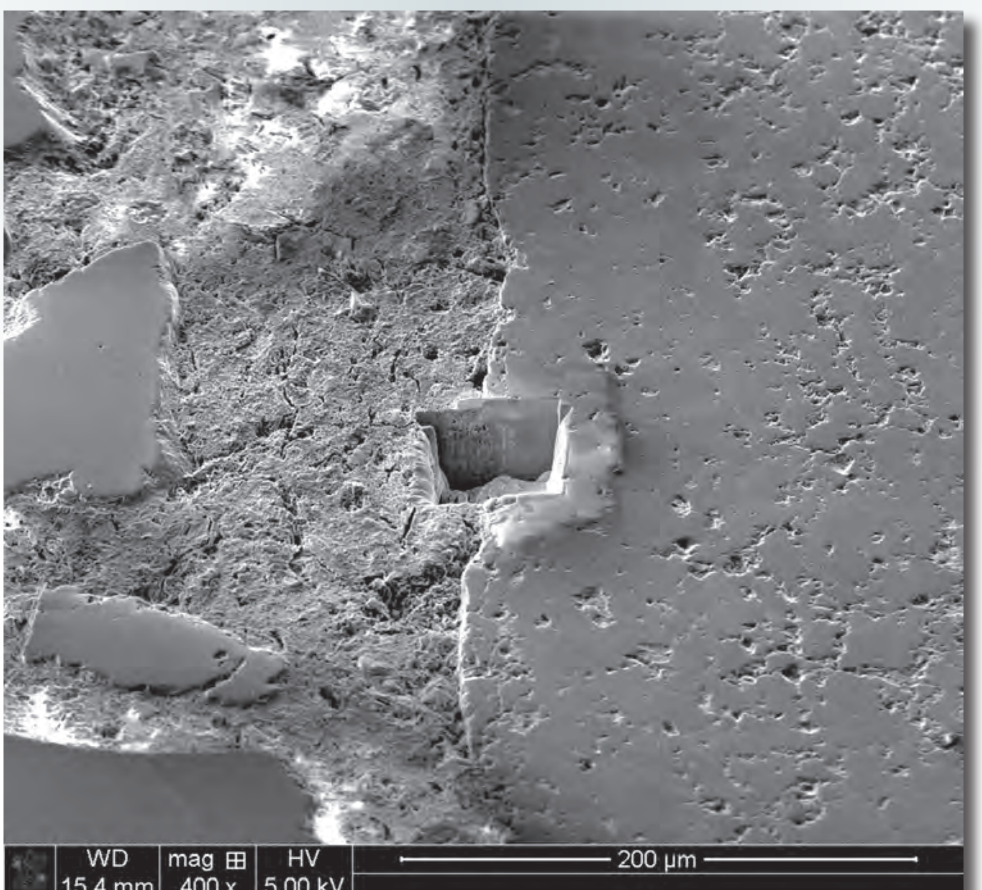
An example of unirradiated hardened concrete recently was investigated at Oak Ridge National Laboratory to determine the feasibility of using FIB-SEM. Figure 3 shows an SEM image of the hardened concrete specimen prepared using FIB. Initial FIB-SEM work shows the perseveration of the interface between the fine-grained aggregate and the cementious matrix, with a change in the porosity of the matrix near the interface. Serial sectioning also could be utilized to gain three-dimensional, structural information of the local pore structure or microcracking near the interface. Additional techniques (such as energy-dispersive x-ray spectroscopy) could be utilized in conjunction with FIB-SEM to provide both structural and compositional information at the interface. The small volume required for such analysis is beneficial for correlating with the small-scale mechanical testing currently being investigated. Further studies on the applicability of the technique to irradiated concrete are being evaluated.

Establishing International Research Partnerships to Understand and Bound the Potential Damage

Understanding the effects of radiation on concrete is important to determining long-term or extended operating performance of concrete structures in existing nuclear power plants; therefore, it is an international endeavor. Not surprisingly, this issue is being addressed by research organizations and utilities across the globe. In the last year, the Materials Aging and Degradation Pathway 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.

For example, multiple discussions (most recently last November in Tokyo) have been held with Japanese research organizations led by the Mitsubishi Research Institute. The focus has been on opportunities to review data from gamma irradiation studies of prototypical concrete and to provide suggestions for additional neutron irradiation experiments underway at the Halden Reactor

Figure 3. A low magnification scanning electron microscopy image showing the cross-sectioning capabilities of focused ion beam scanning electron microscopy and cross-section view of the matrix-aggregate interface of hardened concrete.



Project, Institute for Energy Technology, and JEEP-2 reactor facility. In June 2013, discussions were initiated with the VTT Technical Research Center and Fortum Corporation in Helsinki. The focus was on opportunities to share resources and data, as well as to participate in an international effort to evaluate the potential for radiation-induced degradation and to develop a better understanding of the effects of radiation on structural concrete. Finally, discussions also were initiated in June 2013 with the Research Centre Rez, Czech Republic, concerning opportunities to perform accelerated neutron studies at the LVR-15 reactor and testing at the Czech Technical University in Prague.

Based on these discussions, the direction of this research effort and international cooperation will be to (1) define the upper bound of the neutron and gamma dose levels expected for extended operation, (2) determine the effects of neutron and gamma irradiation, as well as extended time at temperature on concrete, (3) evaluate opportunities to harvest and test irradiated concrete from international decommissioned nuclear power plants (such as Zion in the United States and Zorita through the National Safety Council of Spain), (4) evaluate opportunities to irradiate prototypical concrete under accelerated conditions to neutron and gamma dose levels to establish a conservative bound and/or to share data obtained from different flux, temperature, and fluence levels, (5) develop cooperative test programs to improve confidence in the results from the various concretes and research reactors, and (6) further the understanding of the mechanisms of irradiation degradation in concrete. Clearly, international cooperation will provide 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 very time and cost-efficient manner.

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Session Highlighting Recent LWRS Program Accomplishments at November American Nuclear Society Meeting

A special session highlighting the recent technical accomplishments in the LWRS Program is scheduled for the November American Nuclear Society meeting. The session will take place on Monday, November 11, 2013, beginning at 3:00 p.m., in the Senate Room of the Omni Shoreham Hotel. The pathway leads for the LWRS Program chose the topics and presenters for this session, which are as follows:

- Concrete Aging and Degradation in Nuclear Power Plants: LWRS Program R&D Progress Report (Igor Remec, Oak Ridge National Laboratory)
- Crack Initiation Behavior of Neutron-Irradiated Model and Commercial Stainless Steels in High-Temperature Water (Kale Stephenson, University of Michigan)
- Applying the Results of LWRS Research in Hybrid Control Room Development (Matt Gibson, Duke Energy)
- New Methods and Tools to Perform Safety Analysis within Risk-Informed Safety Margin Characterization (Diego Mandelli, Idaho National Laboratory)
- Station Blackout: A case study in the interaction of mechanistic and probabilistic safety analysis (Curtis Smith, Idaho National Laboratory)
- Out-of-Pile Characterization and Testing of Joined Cylindrical Components for SiC-Based Nuclear Fuel Cladding (Hesham Khalifa, General Atomics).

We hope to see you at the LWRS Program's special American Nuclear Society session in November!

Applying the Results of LWRS Program Research in Hybrid Control Room Development




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Incremental control room modernization is one of the research and development activities supported by the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway. This research and development is especially well suited to enable engineering, verification, and validation activities needed to achieve a gradual replacement of aging analog systems with modern digital instrumentation and control systems. This article highlights the process undertaken to assist Duke Energy with control room modernization, as well as lessons learned in workshops with reactor operator crews.

A Graded Approach to Long-Term Modernization

Supporting long-term operation of nuclear power plants is the key focus of the LWRS Program. Within the LWRS Program, the Advanced II&C Systems Technologies Pathway works closely with utilities to address the challenges associated with incremental control room modernization. Utilities have expressed concerns about the sometimes costly and lengthy process of a license amendment that might be required by the U.S. Nuclear Regulatory Commission for control room replacement (Joe et al. 2012). Beyond the potential licensing delays for significant upgrades, there is a potential for added plant downtime for partial control room replacements. Many utilities have adopted a piecemeal approach, replacing individual instruments as needed (Fink et al. 2004). Because it emphasizes like-for-like replacement of individual analog instruments with digital equivalents, this approach fails to leverage the additional capabilities of new digital instruments and control technologies.

The Advanced II&C Systems Technologies Pathway conducts research to address critical gaps needed to sustain instrumentation and control systems for long-term plant operation. A project on hybrid control room modernization demonstrates methods for integrating human factors, instrumentation and controls, and other engineering work disciplines that are needed to implement modern digital instrumentation and control technologies in place of analog technologies. An objective of these efforts is to support the introduction of technologies that improve reliability, safety, or maintainability over legacy systems. The goal is to ensure that incremental control room modernizations deliver meaningful improvements in performance through their implementation.

For example, a digital control system (DCS) replacement for a legacy analog system might feature digital equivalents of the former instrumentation and control. It also may go beyond this like-for-like capability to include displays and functionalities that enable control room staff to do the following:

- Access important trend information to better anticipate and manage emerging conditions
- Perform calculations that would otherwise be performed manually
- Access prioritized alarm lists to more quickly respond to plant transients
- Obtain helpful checklists that are used to augment paper procedures
- Automate sets of actions that previously were performed manually.

Not all of these improvements can be done within a plant's existing licensing basis; however, such modernizations may improve plant and operator performance to such a degree as to justify the efforts of a regulatory license amendment. The LWRS Program aims to provide systematic studies and evaluations of such technologies prior to implementation in nuclear power plant main control rooms.

To develop and evaluate new control room technologies and methods for their implementation, the LWRS Program has established a full-scope, full-scale control room simulator in its Human System Simulation Laboratory (HSSL), located at the Idaho National Laboratory. The full-scale simulator (see Figure 4) is based on glasstop simulator hardware. The glasstop simulator displays support the display of and operator interaction with simulated analog instruments via touchscreen overlays. A full-scope simulator model drives the panels, which is capable of displaying the full-front panels of contemporary main control rooms at nuclear power plants.

A new DCS is being introduced into the control room of the H.B. Robinson Nuclear Plant. The DCS will upgrade the legacy plant process computer and emergency response facility information system. In addition, the DCS will replace an existing analog turbine control system with a display-based system. In support of upgrading these systems, several workshops were conducted with reactor operators from Duke Energy plants. The purpose of the workshops was to perform a human factors assessment of operators to determine where the DCS might optimize performance.

As part of the systems engineering process for upgrades, human factors engineering focuses on the role and function of humans as key elements in the industrial process. The human-centered analysis of systems and operations integrates the three elements of (1) functional requirements, (2) function allocation, and (3) task analysis to establish design requirements for the control room in nuclear power plants. The functional requirements are determined by systems engineers at the plant, while function allocation and task analysis require human factors expertise seldom found in-house at the plant. Idaho National Laboratory human factors researchers involved in the LWRS Program provided this expertise and conducted a series of workshops using HSSL.

"Don't let your precision exceed your accuracy!" exclaims Matt Gibson, the project coordinator from Duke Energy. "A lot of people struggle with the particular details, but miss the big picture. Human factors is a science that converges to a particular level of precision, but once you reach a certain point, stop and move forward. These workshops in HSSL allowed us to get the answers we needed quickly and accurately."

Function Allocation Workshop

A function allocation workshop was held with operators from several Duke Energy nuclear power plants to develop and validate requirements for the new digital turbine control system. In preparation for the function allocation workshop, plant scenarios were developed that included use of the existing plant process computer and emergency response facility information system that will be replaced by a DCS. The function allocation workshop was conducted

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Figure 4. The Light Water Reactor Sustainability Program's glasstop simulator at the Idaho National Laboratory.

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in November 2012 on the glasstop simulator at HSSL (see Figure 5).

The reactor operators agreed that the simulator achieved the workshop objectives (i.e., identifying key functions and opportunities for refining the allocation of those functions to either operator or automation). In addition, the workshop was instrumental in developing and testing methods to conduct function allocation activities with the multi-disciplinary teams that are needed to meet the objectives of NUREG-0711 (O'Hara et al. 2012), which is the human factors engineering process that guides the regulatory review of operator technologies.

Task Analysis

Task analysis defines the information, control, and human performance requirements for various operational scenarios. A task analysis workshop also was conducted with Duke operators at HSSL and at nuclear power plants to identify and analyze the specific tasks that need to be performed by operators to accomplish the requirements of specific and general operating conditions. The ultimate aim of the task analysis phase of the control room upgrade is to develop high-level, sequential descriptions of operations that must be carried out to fulfill the functions of a particular system. Information from the scenario walkthroughs and debriefs was recorded in operational sequence diagrams that map the operators' interactions with the various systems in the control room.

Findings From the Workshops

The function allocation and task analysis workshops identified new functionality that would be advantageous to operators in modernization of the plant process computer and turbine control system. Example new functions that were identified included the following:

- Procedure support displays—can be called up to support operators walking through particular commonly used or complex procedures (e.g., a display to bring up plant parameters required to step through the checklists in an initial emergency operating procedure, E-0)
- Automated calculations—currently have to be calculated manually by operators from separate and sometimes distally located indicators in the current control room configuration (e.g., leak rate calculation)
- Shot clocks—help the operator keep track of time required for continuous action steps, including multiple simultaneous continuous actions
- Prioritized alarms—help the operator to focus on the most safety critical tasks at hand.

Additionally, work scope has been established to use HSSL in support of the remaining phases of NUREG-0711 (i.e., design, verification and validation, and implementation). Initial work is being conducted to plan DCS replacements (see Figure 6) across multiple Duke Energy plants, beginning with the layout of the boards to enable the introduction of DCS displays (see Figure 7). The HSSL virtual boards allow staging of multiple layouts to arrive at the most ergonomic placement of displays. Software prototypes of the DCS can be tested on HSSL using operator-in-the-loop studies.

As the phases of the human factors engineering process are completed, they are being disseminated as joint reports with the Electrical Power Research Institute. The joint LWRS

Figure 5. Observation and debriefing in the simulator.

Figure 6. Matt Gibson of Duke Energy works with Advanced II&C Systems Technologies Pathway researchers to plan for the placement of new digital control system displays in an existing control room.

Figure 7. Example of modifications to the main control boards that enable the introduction of new digital displays.

Program/Electrical Power Research Institute Long-Term Operations Program reports will provide relevant document templates, step-by-step guidance, and lessons learned for control room modernization across the U.S. nuclear industry. The first joint branded report in this series is anticipated to appear in early fiscal year 2014. "The HSSL provides the ability to rapidly develop prototype control-room modifications, get early feedback from control room operators, and test new designs with realistic plant scenarios before the designs are built," said Joseph Naser, Electrical Power Research Institute project manager and technical executive. "This will allow the designs to effectively and reliably meet the goals of the plant owner and will reduce the cost and time to implementation."

Conclusions

This work is ongoing; however, a key use has emerged for the simulator. HSSL can serve as a standard test bed for control room modernization across the industry. The simulator incorporates a plant-neutral configuration, based on glasstop panels that mimic current control panels. This design is especially suited to modernization efforts as plants make the transition from analog panels to hybrid control rooms, incorporating legacy analog and new DCS technology. This capability fills a crucial gap in existing research simulators, which primarily have been used to develop workstation-based next generation control rooms. The transitional, hybrid phase that is realistic in the U.S. nuclear industry has not benefitted from a dedicated research facility prior to HSSL.

HSSL has already been successful in its short existence. Future work will continue with the current approach, by doing the following:

- Facilitating innovative technology solutions (e.g., advanced alarm management systems and advanced DCS visualizations) to ensure that control room upgrades go beyond current technological limitations while maintaining current high safety and reliability
- Documenting the complete control room modernization process to augment existing guidance and provide a template that can benefit the utility and regulatory communities
- Incorporating additional plant simulator models to ensure that lessons learned from control room modernization properly generalize across different reactors and plants in the United States
- Deploying plant models from additional simulator vendors to ensure maximum reconfigurability and flexibility of HSSL.

The research and development efforts being undertaken by the LWRS Program in this area are being used to establish a systematic and rigorous approach to human factors engineering in the control room modernization process. The methods and demonstrations of research are being used by utilities to develop and implement plans for gradual new technology replacement of legacy control room instrumentation in the nuclear industry.

Matt Gibson notes, "People have got to get over the idea that human factors is something that takes years of effort. This project is a perfect example of how human factors

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can quickly become an indispensible part of the upgrade process. It's doable, it's scalable, and it adds value."

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Recent LWRS Program Reports

Materials Aging and Degradation

- ***Analysis of Deformation Mode Changes in Irradiated Materials using Bend Tests and Finite Element Modeling***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ORNL-TM-2013-217_Deformation_Mode_Changes_M3LW-13OR0402024.pdf
- ***Summary of Large Concrete Samples***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ORNLTM-2013223_ORNLConcreteSamples_M3LW-13OR0403034.pdf
- ***Modeling Strategy to Assess Radiation Induced Segregation and Phase Stability in Austenitic Steels in Light Water Reactors During Extended Service***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/UW_UT_Milestone_M3LW-13OR0402054.pdf
- ***Report on Small-Angle Neutron Scattering Experiments of Irradiated Reactor Pressure Vessel Materials***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ORNL-TM-2012-630_Small-Angle_Neutron_Scattering_M3LW-13OR0402012.pdf

Advanced II&C Systems Technologies

- ***Installation of Halden Reactor Project Digital Interface Prototypes in the Human Systems Simulation Laboratory***
  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/INL-EXT-13-29039_Halden_M3LW-13IN0603023.pdf

Risk-Informed Safety Margin Characterization

- ***RELAP-7: Demonstrating the Integration of Two-phase Flow Components for an Ideal BWR Loop***
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Advanced Light Water Reactor Nuclear Fuels

- ***Status of Silicon Carbide Joining and Irradiation Studies***
  https://lwrs.inl.gov/Advanced%20Light%20Water%20Reactor%20Nuclear%20Fuels/ORNLTM-2013273_SiC_Joining_Report_M3LW-13OR0504073.pdf
- ***Evaluation of CVD and CVC SiC Reactivity towards UO_2***
  https://lwrs.inl.gov/Advanced%20Light%20Water%20Reactor%20Nuclear%20Fuels/CVC-CVD-SiC_Out-of-pile_Testing_M3LW-13IN0502045.pdf

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