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# Industry Application: External Flooding Analysis using the Risk-Informed Safety Margin Characterization Methodology

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Risk-Informed Safety Margin Characterization Pathway

### *Introduction*

The Risk-Informed Safety Margin Characterization (RISMC) Pathway is modernizing nuclear power safety analysis (i.e., tools, methods, and data); implementing state-of-the-art modeling techniques; taking advantage of modern computing hardware; and combining probabilistic and mechanistic analyses to enable a risk-informed safety analysis process. The modernized tools provide an improved understanding of safety margins and the critical parameters that affect them, thereby aiding plant operators in their efforts to maintain the current high levels of safety in the commercial nuclear power fleet.

The computational power available to safety analysis has grown remarkably and is expected to continue to increase. The improvements described in this article are made possible by computational improvements and the inclusion of multiple types of physics that interact and (potentially) impact safety.

Technical exchange meetings held in 2014 identified high-priority items that industry and the U.S. government would be interested in pursuing related to the development of safety activities. The results of the prioritization and selection of "Industry Applications" to be investigated under the RISMC Pathway are documented in external reports (Szilard and Smith 2014, Szilard et al. 2014). Given that hazards external to a nuclear facility may negatively impact a variety of systems, structures, and components (SSCs) from direct damage (e.g., failure during a fire) or indirect damage (e.g., consequential failure from a flood following a pipe break), the Industry Application related to external hazards was identified as one of the high-priority activities. This current application of the RISMC methodology specific to flooding hazards is described in this article.

To evaluate the possible impact of hazards on the existing fleet of nuclear power plants, the RISMC Pathway aims to provide insight for decision makers through a series of plant dynamics simulations for different initial conditions. Research and development looks at challenges to a hypothetical pressurized water reactor, including (1) a potential loss of offsite power, followed by possible loss of all diesel generators (DGs); (2) an earthquake-induced station blackout; and (3) a potential earthquake-induced tsunami flood. The following codes were used to perform the analysis: a thermal-hydraulic code (i.e., Reactor Excursion Leak Analysis Program-7 [RELAP-7]); a flooding simulation tool (i.e., NEUTRINO); and a stochastic analysis tool (i.e., Risk Analysis and Virtual Control ENvironment [RAVEN]). Using RAVEN, multiple RELAP-7 simulation runs were performed by changing the status of specific system/components of the model to reflect specific aspects of different scenarios, including both failure and recovery of

> ***External Hazards***
>
> A class of initiating events for a nuclear facility that originates external to the plant.

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critical components. The simulation employed traditional statistical tools (such as Monte-Carlo sampling) and more advanced machine-learning based algorithms to perform uncertainty quantification to understand changes in system performance. Qualitative and quantitative results obtained gave a detailed picture of the issues associated with potential accident scenarios. These types of insights can provide useful information for decision makers to perform risk-informed margins management.

### *Example of the Flooding Analysis*

The following steps were taken to perform a flooding analysis using the RISMC method:

- *Initiating event modeling:* modeling characteristic parameters and associated probabilistic distributions of the event considered
- *Plant response modeling:* modeling of plant system dynamics
- *Components failure modeling:* modeling of specific SSCs that may randomly change status (e.g., fail to perform specific actions) due to the initiating event or other external/internal causes
- *Scenario simulation:* when all modeling aspects are complete (see previous steps), a set of simulations can be run by randomly sampling the set of uncertain parameters.

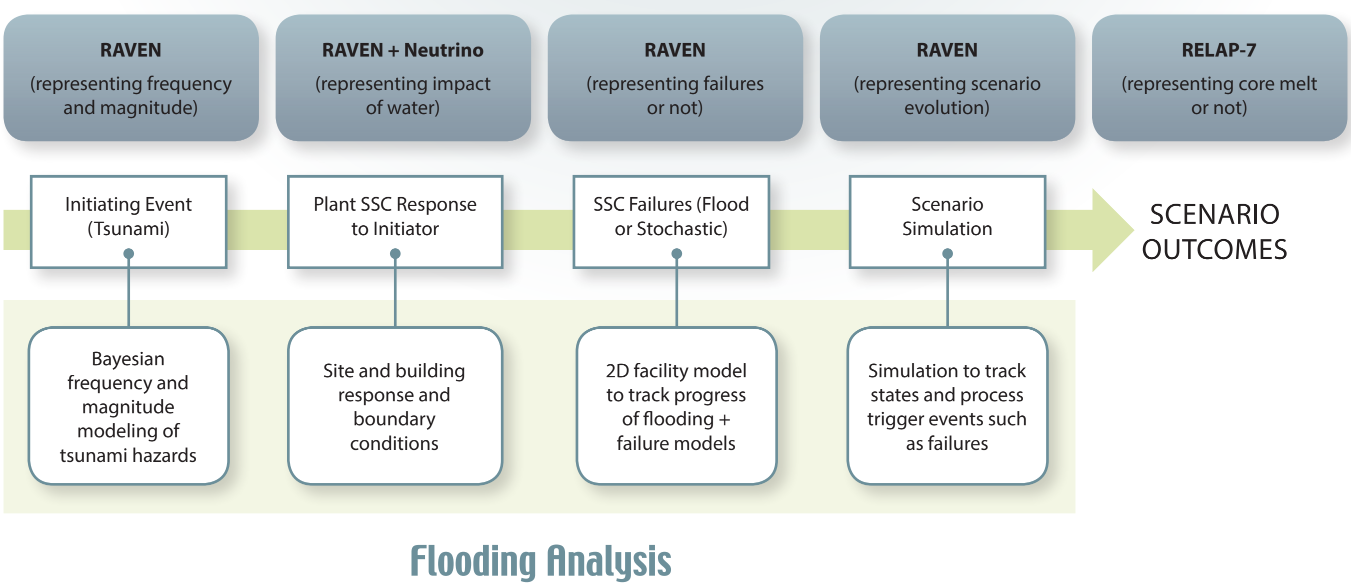
These steps are shown in Figure 1.

Scenario simulation generates a set of statistical information (e.g., the probability of core damage). Also of interest is determining the limit surface (i.e., the boundary between failure and success). A generic three-dimensional (3D) facility model (see Figure 2) was created and used to simulate various tsunami-flooding scenarios. For initial testing, a single reactor unit was used and includes the turbine building, reactor building, offsite power facilities and switchyard, and DG building.

Other onsite buildings were included in the model to realistically represent the flow of a tsunami through the site; however, these buildings were not evaluated for internal flooding in this initial demonstration. Further, for this demonstration, all objects are fixed rigid bodies. Future analysis will explore the possibility of moving debris (caused by the flood) and possible secondary impacts due to this debris.

To mimic a tsunami entering the facility, a bounding container was added around the perimeter of the model (not shown in Figure 2) and for the ocean floor (this keeps the water from leaving the simulation). Then, over 12 million simulated fluid particles were added for the ocean volume. A wave simulator mechanism was constructed by having a flat planar surface that moves forward and rotates, pushing the water and creating a wave in the fluid particles (similar to how tsunami research facilities actually create scale-model waves). Once the wave is started, the fluid solver handles all of the remaining

***Figure 1. RISMC steps followed to perform the flooding analysis.***



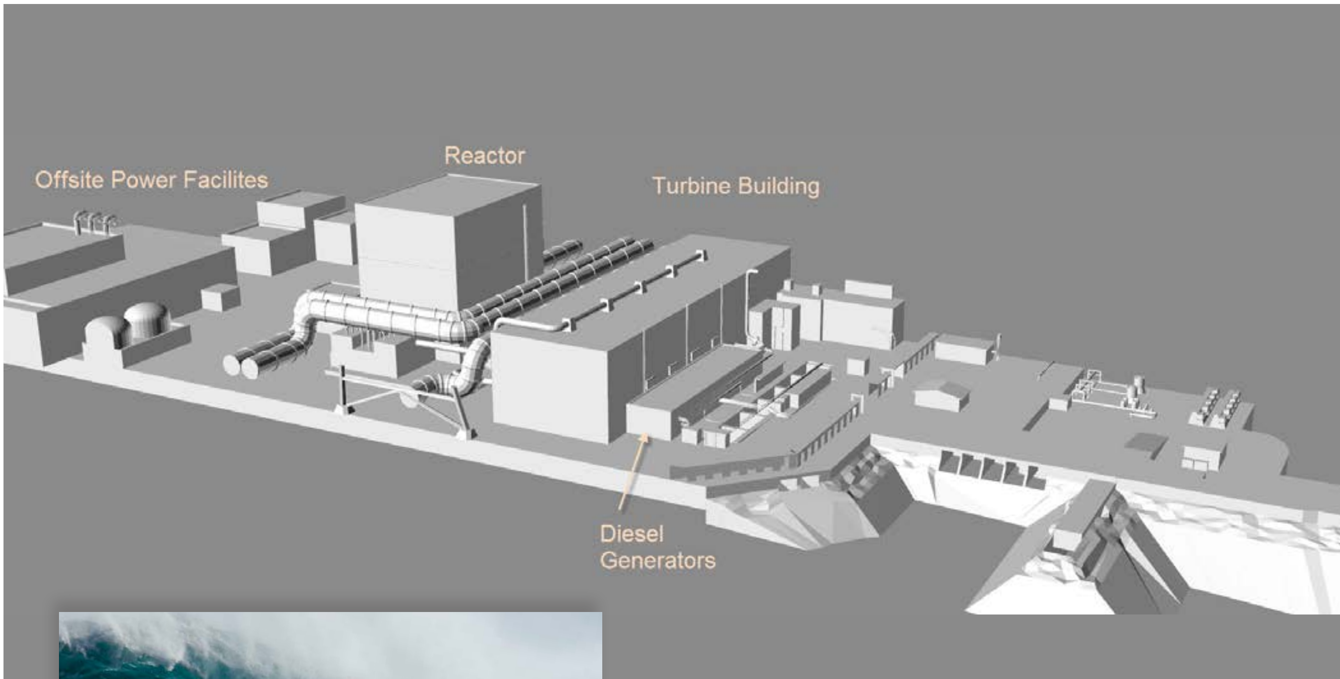
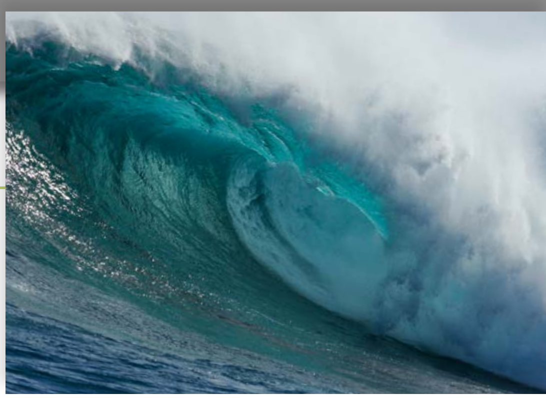



Figure 2. 3D plant model developed to simulate flooding.



calculations to simulate the physics of the moving wave through the facility. Further, various wave heights can be generated through minor parameter adjustments to the movement of the wave generator.

Several different approaches simulate and optimize fluid movement (Wikipedia 2015), each having different advantages and purposes. To achieve the most realistic and accurate results, a smooth particle hydrodynamics-based solver called NEUTRINO was used. NEUTRINO also factors in advanced boundary handling and adaptive time stepping to help increase accuracy and calculation speed. Most simulations for the RISMC flooding demonstration were performed with a single workstation using seven processors (each running two calculations in parallel). Each tsunami simulation took approximately 3 minutes per frame, with a total run time ranging from 75 to 90 hours, depending on how many frames were needed for the simulation. This is a reasonable amount of time given the calculation that is taking place; however, parallel methods are being investigated to improve the speed of analysis.

As the particles of the simulation move, they interact with the rigid bodies of the 3D model. The simulated fluid flows around buildings, splashes, and interacts in a similar manner to real water. Software measuring tools can also be added to the simulation to determine the fluid contact information, water height, and flow rates into openings at any given time in the simulation. This dynamic information can be used in two ways: (1) a static success/failure of components/structures depending on wave height or (2) a dynamic flow result based on time for use in a more detailed analysis (e.g., to start running an internal flooding calculation representing the water flow into a building).

Several simulations were run at different wave heights. Fluid penetration into the site is measured for each of the simulations to determine at what height the different systems fail inside different parts of the facility. For this specific case, venting is monitored for DGs and offsite power structures. As shown in Figure 3, fluid particles are entering both air intake vents for an 18-meter wave. Evaluating this scenario in more detail, we can determine that at simulation time (or frame) 1,280, DG1 fails from particles entering the vents and DG2 fails at frame 1,375.

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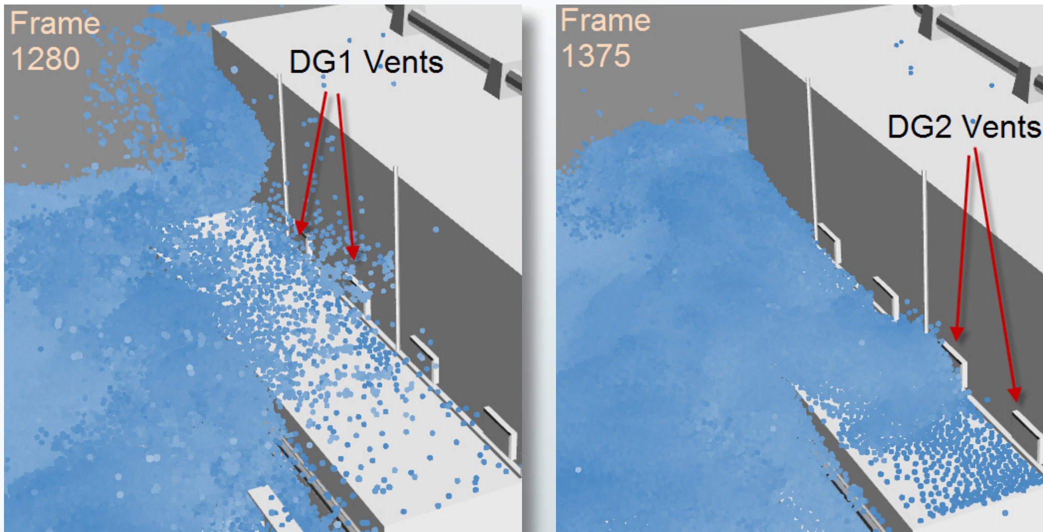


*Figure 3. Example of water entering vents during a simulation at different points in time.*

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A series of simulations were performed using the NEUTRINO code on the 3D plant model to measure plant response for several wave heights in the 0 to 30-meter range. The DGs tended to fail more with smaller waves (i.e., above a minimum value) than the power grid structures, because the DG building is closer to the ocean shore and the air intake vents face the wave directly. If the simulated wave was greater than 18 meters, water entered into both of the DGs air intake, while the power grid switchyard was flooded only for a wave height greater than 30 meters (see Table 1).

Coupling the fluid physics represented by the tsunami with the RAVEN-generated scenarios provides the ability to describe the plant's operational boundary conditions. During the simulation, these conditions are provided to the RELAP-7 code to evaluate the plant's thermal-hydraulics. The safety margin is determined directly as a function of the scenario evolution; for example, the case shown in Figure 4, where the safety limit was exceeded during an extreme scenario (i.e., 22.4-meter tsunami). This simulation process is repeated until an adequate number of simulated runs are captured (see Smith et al. [2014] for additional details).

### Summary

This article summarizes the series of steps needed to evaluate a RISMC Industry Application case study for flooding using RAVEN, NEUTRINO, and RELAP-7. Analysis was conducted by modeling the pressurized water reactor system dynamics using the RELAP-7 code and the flooding scenario using the NEUTRINO code.

*Table 1. Status of the DGs and power grid switchyard for different tsunami heights.*

| Wave Height (meter) | DG1 Status | DG2 Status | Offsite Power Switchyard Status |
|---|---|---|---|
| Less than 17 | OK | OK | OK |
| 17 to 18 | Failed | OK | OK |
| 18 to 30 | Failed | Failed | OK |
| Greater than 30 | Failed | Failed | Failed |

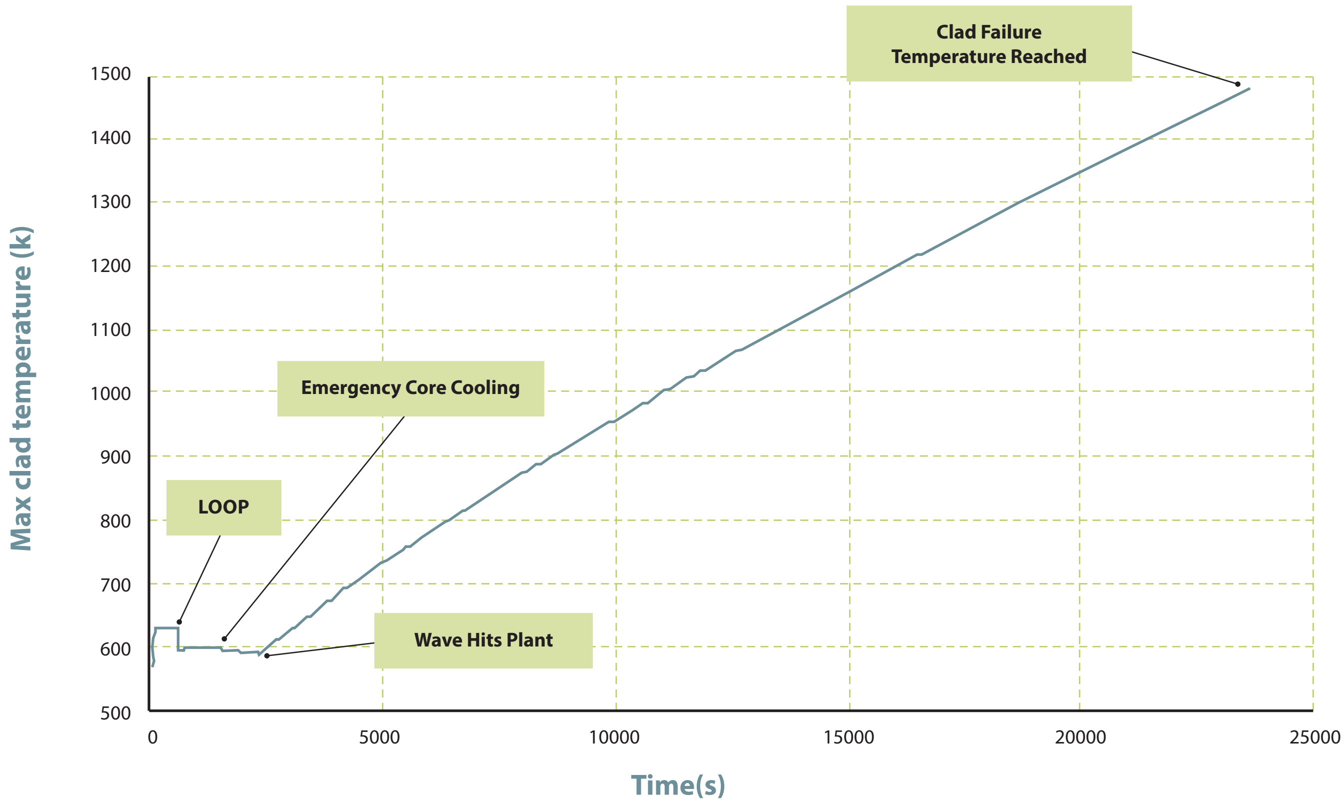


***Figure 4. Example of a sample scenario leading to core damage due to a 22.4-meter high wave hitting the plant at about 30 minutes after loss of offsite power (LOOP). When the wave hits the plant, because its height is above 18 meters, the DGs are disabled and recovery times are past the point of core damage.***

As part of the more detailed analysis found in Smith et al. (2014), focus was on steps for completing the analysis so that risk information could be obtained. This information could be used to perform risk-informed margins management decisions. For the hazard of tsunami flooding, potential modifications to be considered include the following:

- Increase the wave protection wall to reduce the flooding level in the plant. This will act on the fraction of the wave height distribution that causes DG failure.
- Improve AC emergency recovery procedures. This action acts directly on either the DG or power grid recovery distribution (i.e., a lower DG or power grid average recovery time).
- Move some or all DGs to a non-flood prone area of the plant site.
- Improve bunkering of the DG building in order to reduce the likelihood of flood-caused failures.

The RISMC methodology, coupled with the RISMC Toolkit, provides an enhanced approach for supporting engineering decisions concerning complex external events such as flooding.

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# Integrating Plant Experience with Expert Panels and Research Results: The Zion Harvesting Project and the Constellation Joint Demonstration Pilot Project

**Thomas M. Rosseel**

Materials Aging and Degradation Pathway

It is well understood that components and structures in a nuclear power plant must withstand a very harsh operating environment, including extended time at temperature, neutron irradiation, stress from operational loads, and corrosive media. Moreover, extending reactor service beyond 60 years will increase those demands and possibly introduce new modes of degradation (Busby 2015). Although the numerous modes of degradation are complex and vary depending on location and material, understanding and managing materials degradation is key for continued safe and reliable operation of nuclear power plants. As noted in the Expanded Materials Degradation Assessment (EMDA) (EMDA 2014), which is a comprehensive evaluation of potential aging-related degradation modes, an important component of understanding materials degradation is the examination of service-aged materials. Two important sources of service-aged materials are the Zion Harvesting Project and the Constellation Joint Demonstration Pilot Project (Busby 2015). These projects are key because access to materials from active or decommissioned nuclear power plants provides an invaluable resource for which there is limited operational data or experience to inform relicensing decisions and assessments of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior.

### *Zion Harvesting Project*

The Zion Harvesting Project, in cooperation with Zion Solutions, LLC, which is a subsidiary of Energy Solutions (i.e., an international nuclear services company), is coordinating selective procurement of materials, structures, components, and other items of interest. These items are procured for the Light Water Reactor Sustainability (LWRS) Program, the Electric Power Research Institute (ERPI), and the U.S. Nuclear Regulatory Commission (NRC) from the Zion Station (a former nuclear generating facility) in support of extended service and current operations of the U.S. nuclear reactor fleet. The Zion Station is a decommissioned, two-unit, Westinghouse, four-loop pressurized water reactor facility, with each unit capable of producing 1,040 MWe. The units were commissioned in 1973, permanently shut down in 1998, and placed into SAFSTOR in 2010 (i.e., a method of decommissioning where a nuclear facility is placed and maintained in a condition that allows the facility to be

***Table 2. Location of cables requested for harvesting from Zion Station Unit 2 and received from Unit 1.***

| Requested Items | Plant Location | Status |
|---|---|---|
| Control Rod Mechanism Power and Position Cables | Inside missile barrier, RPV head (Unit 1) | Received |
| Pressurizer Heater Cabling | Inside missile barrier on the pump deck | Requested |
| Loop Isolation Valve Cabling | Inside missile barrier on the pump deck | Requested |
| Accumulator Discharge Motor-Operated Valve Cabling | Outside missile barrier, lower level of containment | Requested |
| Instrumentation Cables | Instrument racks outside missile barrier lower level of containment | Requested |
| Air-Operated Valve Cabling | Pipe penetration area, intermediate level, outside missile barrier | Requested |
| Instrumentation Cabling Bundle | Inside missile barrier, lower level | Requested |
| Cable Penetration Area | Upper level of containment, outside missile barrier | Requested |
| Steam Tunnel Cabling | Overhead in the steam tunnels | Requested |
| Cable Penetration Vaults | Cable penetration vaults outside containment | Requested |

safely stored and subsequently decontaminated to levels that permit release for unrestricted use). Materials of high interest include low-voltage cabling, concrete core samples, through-wall-thickness sections of the reactor pressure vessel (RPV), and other SSCs of interest to researchers evaluating aging management issues (Rosseel et al. 2014).

### *Zion Station Low-Voltage Cables*

One potential knowledge gap identified in the EMDA recommendations (EMDA 2014) includes cables in high radiation zones (i.e., 70 Mrad over 80 years) between 45 and 55°C. The LWRS Program and NRC have requested 10 sets of low-voltage cables for harvesting (see Table 2 and Figures 5 through 8) from radiation, thermal, radiation and thermal, and ambient environments from Zion Station Units 1 and 2. Analysis of these components will help inform several knowledge gaps identified in the EMDA. Selection of the type and location of cables was based on potential damage caused by the plant service environment and observations during site visits. The first batch (i.e., six sets of 25 to 30-ft lengths of control rod drive mechanism power and position indicator cables) was harvested in 2012. Testing of NRC cables has begun at the National Institute of Standards and Technology. The second batch of approximately nine sets of 60 to 100-ft length cables should be harvested in late spring 2015. The objective of this work is to understand and predict cable degradation at extended lifetimes. Research is focused on (1) validating predictive models (based on accelerated aging studies) with empirical data obtained from field-aged materials and (2) providing greater confidence in the performance of cables during an accident, with measurable indicators in lieu of relying on the current methodology of calculating service life based on environmental monitoring. For these reasons, this effort is focused on cables that have been exposed to thermal and radiation environments (i.e., in-containment cables). Cables in high thermal environments and those in a benign environment (i.e., cable spreading room) may provide a baseline for separating the effects of radiation and high thermal environments.

### *Zion Station Concrete Core Samples*

The EMDA report identified an urgent need for developing a consistent knowledge base on irradiation effects in concrete (EMDA 2014). Unfortunately, much of the historical mechanical performance data on irradiated concrete (NUREG 2006) does not accurately reflect typical radiation conditions in nuclear power plants or conditions out to 60 or 80 years of radiation exposure (EMDA 2014). To address these knowledge gaps, the EPRI Long-Term Operations and LWRS Programs are working together to better understand radiation damage as a degradation mechanism. Research in this joint effort has focused on (1) defining the upper bound of the neutron and gamma dose levels expected in the concrete biological shield for extended operation (80 years of operation and beyond); (2) determining the effects of neutron and gamma irradiation and extended time at temperature on concrete; (3) evaluating opportunities for irradiating prototypical

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***Figure 5. Zion Station Unit 2 cable penetration vault, outside containment.***

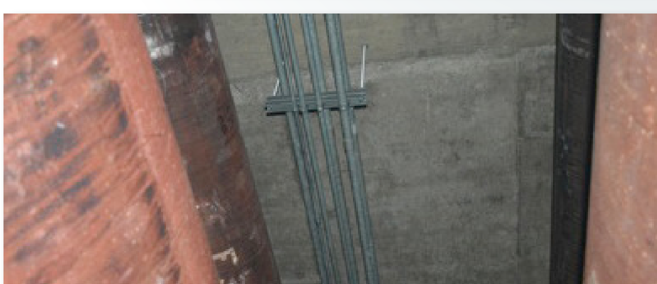

***Figure 6. Zion Station Unit 2, main steam-tunnel cabling in the overhead.***


***Figure 7. Low voltage cabling located near a reactor coolant pump inside Zion Station Unit 2 containment.***

***Figure 8. Zion Station Unit 1, control rod drive mechanism power and position indicator cable bundles.***

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aggregate and concrete under accelerated neutron and gamma dose levels to establish a conservative bound and share data obtained from different flux, temperature, and fluence levels; (4) evaluating opportunities for harvesting and testing irradiated concrete from international nuclear power plants; (5) developing cooperative test programs to improve confidence in the results from the various concretes and accelerated radiation experiments; and (6) establishing an international collaborative research and information exchange effort to leverage capabilities and knowledge (Rosseel et al. 2014).

The LWRS Program is working with Zion Solutions to harvest three sets of concrete cores from eight to ten locations at the Zion Station. The concrete cores will be characterized for changes in mechanical properties (e.g., strength, modulus, hardness, and density) and microstructural features at the onset of possible aggregate swelling and cement paste cracking. These tests are designed to develop a better understanding of and ability to predict concrete degradation at extended lifetimes. Research will be focused on (1) validating predictive models based on accelerated aging studies, with empirical data obtained from field-aged concrete in radiation and thermal environments; and (2) evaluating concrete radiation gradients (e.g., the concrete biological shield) to investigate the changes in properties as a function of the level of radiation. With the addition of concrete from ambient or controlled environments (e.g., the cable spreading room), it may be possible to separate the effects of radiation and thermal environments.

### *Zion Station Electrical Components*

As requested by the Fire Protection Group within NRC's Office of Nuclear Regulatory Research, an approximately 8-ft section of 4-kV bus bar, with the enclosure and internal supporting isolator hardware, was harvested from the Zion Station's 4-kV switchgear room. The desired section (shown in Figure 9) consists of an approximate 2-ft elbow and approximately 6-ft of the horizontal piece of the bus bar. This unit was shipped to the National Institute of Standards and Technology in March 2014 for studies evaluating burn patterns caused when the unit is energized beyond normal operating conditions.

### *Zion Station Reactor Pressure Vessel Segments*

A potentially life-limiting component in light-water reactors (LWRs) is the RPV because replacement of the RPV is not considered a viable option (Rosseel et al. 2012). Researchers studying the effects of radiation on RPV materials have long been interested in evaluating service-irradiated materials to validate physically informed correlations of transition-temperature-shift predication models (Eason et al. 2013). For those reasons, the LWRS

***Figure 9. Eight-foot section of 4-kV bus bar, with the enclosure and internal supporting isolator hardware, harvested from the Zion Station's 4-kV switchgear room.***

Program is focused on acquisition of two segments of the Zion Station Unit 2 RPV, cutting the segments into blocks from the well-characterized beltline and upper vertical welds and base metal, and machining those blocks into mechanical (i.e., Charpy, compact tension, and tensile) test specimens and coupons for microstructural (i.e., transmission electron microscopy, atom probe tomography, small angle neutron scattering, and nano indention) characterization. Specifically, the objective is to harvest one segment of the SA-1769 beltline weld with a WF-70 vertical weld and an adjoining section that includes a segment of the SA-1769 beltline weld but no vertical weld (Figure 10).

The Zion Station Unit 2 RPV is composed of the head, nozzle ring section, two ring or shell sections (with hemispherical plates that have two vertical welds), and a bottom plate. Based on information provided by Zion Solutions, the vessel will be cut into 22 segments over four levels using an oxy-propane torch. Level 2 will be cut into eight 45-degree segments of 157.5 inches in height, 72.9 inches in length as measured from end to end of the outer diameter, and 8.8 inches thick, including a 3/16-inch stainless steel cladding on the internal surface. Each piece will weigh approximately 28,000 pounds. These segments will include most of the intermediate and lower shells, and each segment will include a portion of the beltline weld. Two of the segments will include the WF-70 vertical weld of the intermediate shell, two will include the WF-29 vertical weld of the lower shell, and four will not include any vertical weld. Because the Level 2 cuts are expected to occur from just below the circumferential weld of the upper and intermediate rings and just above the radial guides, there will be no thermal damage to the beltline weld or heat-affected zone from the horizontal cuts. Based on the current schedule, Zion Solutions expects to generate the desired segments in spring 2015 (Rosseel et al. 2015).

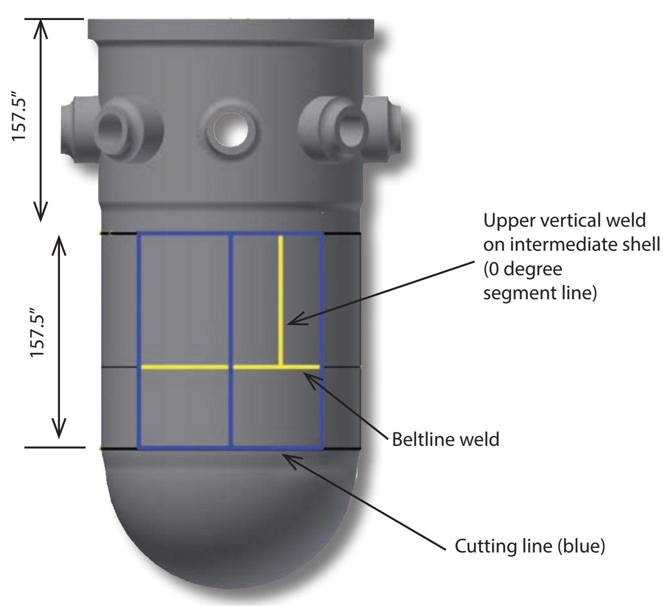
The two RPV segments harvested from Zion Station Unit 2 include the SA-1769 beltline weld with a WF-70 vertical weld and an adjoining section that includes a section of the SA-1769 beltline weld, but no vertical weld will be cut into two types of blocks of varying lengths. The two types are designated as "C" and "F". The "C" block will be used to machine Charpy V-notch, tensile, and coupon specimens, and the "F" block will be used to machine compact tension specimens for fracture toughness testing. Prior to cutting the "C" and "F" blocks and machining the test specimens, the location of the center line of the welds will be identified using chemical etching techniques (Rosseel et al. 2015).

Data from RPV surveillance specimens containing similar SA-1769 and WF-70 weld materials are available in literature for a comparison of hardening and changes in fracture toughness and microstructure (McCabe et al. 2000, Terek et al. 1989, Carter et al. 2001). Moreover, the harvested segment (containing the beltline weld only) is expected to have the highest radial fluence and will also provide a source of base metal. Access to service-irradiated RPV welds and base metal will allow through-wall attenuation studies to be performed, which will be used to assess current radiation damage models (Rosseel et al. 2012).

### *Constellation Joint Demonstration Pilot Project*

The Constellation Joint Demonstration Pilot Project is a multi-party venture among the LWRS Program, EPRI, and the Constellation Energy Nuclear Group (Exelon). The project utilizes two of Constellation's nuclear stations, R. E. Ginna and Nine Mile Point 1, for research opportunities to support subsequent license renewal and aging management due to their long operating history. Specific areas of joint research have included installation of equipment for monitoring containment rebar and concrete strain, containment temperature, microstructural analysis of RPV surveillance coupons, and plans for evaluating baffle bolts (Busby 2015). This article describes the work at Ginna; the Ginna reactor is a Westinghouse two-loop pressurized water reactor that began commercial operation in July 1970 and produces about 580 MWe.

***Figure 10. The WF-70 vertical weld with the beltline weld from the segment centered at 0 or 180 degrees and an adjacent segment with only the SA-1769 (weld wire 72105) beltline weld.***




### *Collection and Assessment of the Ginna Reactor Cavity Concrete Temperature*

It is well known that the containment building of a nuclear power plant is affected by a number of factors, including radiation, load, and temperature. During the last operating cycle (from 2012 through 2014), temperature monitoring was performed in the Ginna pressurizer cubicle, with a maximum temperature of 130.7°F recorded. Because this temperature is considered the hottest location in containment, it was confirmed that the temperature of concrete in containment is well within the temperature limits recommended by the American Concrete Institute Code of 150°F for sustained temperature exposure to concrete (LPI Report 2013). This data assessment is important because it helps assess operating conditions for models and mechanistic studies and evaluating operating assumptions for the entirety of the fleet.

### *Tendon Monitoring at Ginna*

A 3-year effort to develop and evaluate an augmented inspection and testing program of the vertical post-tensioned tendons was performed to supplement the industry's understanding of concrete containment structures at long-term operation. Specifically, fiber-optic strain gages and temperature sensors were installed on 20 tendon shims in 2011. The gages were correlated to the lift-off load during tendon lift-off tests performed in April 2011. The lift-off test is the standard method for evaluating the conditions of containment tendons. Data collection was initiated in April 2011 and concluded in June 2014. The response of the tendons and concrete during the pressurization test was consistent with analyses that were performed. The change in tendon loads from 2011 to 2014 demonstrated that those loads were acceptable at the end of the monitoring period. Moreover, the demonstration project showed that the fiber-optic sensors are generally stable and can provide meaningful online data of the loads in the tendons. Although several gages showed behavior that appeared to be drift and should be evaluated as additional data becomes available, no tendon breaks were detected during this study (LPI Report 2014).

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### *Microstructural Analysis of Ginna Reactor Pressure Vessel Surveillance Coupons*

As discussed in the EMDA report and elsewhere (EMDA 2014, Sokolov et al. 2014), RPV has sufficient fracture toughness that failure is implausible under any postulated condition, including pressurized thermal shock in a pressurized water reactor. However, in the irradiated condition, the fracture toughness of RPV may be severely degraded, with the degree of toughness loss dependent on the radiation sensitivity of the materials. In addition, existing embrittlement predictive models (Eason et al. 2014) and our present understanding of radiation damage are not fully quantitative and do not treat all potentially significant variables and issues, particularly considering extension of operation to 80 years. For these reasons, previously tested Charpy impact surveillance specimens of forging, weld metal, and heat-affected-zones from the Ginna nuclear power plant have been examined with atom probe tomography and small angle neutron scattering (Sokolov et al. 2014).

Five capsules have been removed from Ginna and tested over the course of reactor operation at neutron fluences ranging from 0.58 to $5.8\times10^{19}$ n/cm$^2$. The specimens provided to Oak Ridge National Laboratory were irradiated to 1.69, 3.64, and $5.80\times10^{19}$ n/cm$^2$, with the latest removal of Capsule N after 30.5 effective full power years of plant operation. All received specimens were tested by Charpy impact and, as expected, based on the higher copper (Cu) content of the weld relative to the forging, the weld exhibits significantly greater irradiation-induced shifts. The Ginna weld metal samples contain about 0.23 wt.% Cu and the precipitates (Figure 11) reveal strong enrichment of Cu, as well as nickel, manganese, and silicon. The higher Cu weld also exhibits ultrafine Cu, nickel, manganese, and silicon and phosphorus-enriched precipitates and phosphorus segregation to the dislocations (Sokolov et al. 2014).

Similar behavior was observed in the intermediate and high-fluence weld specimens. The size and number densities of the Cu-enriched precipitates were much smaller in the forgings due to the lower Cu contents. Also, there was a trend for the number densities to increase with higher fluence. A preliminary observation from this and other atom probe tomography experiments is that the irradiation-induced Cu-enriched precipitates from surveillance and test reactor irradiations have nominally the same characteristics. A graph of bulk Cu level in the atom probe tomography samples (atomic %) versus the precipitate number density for the Ginna surveillance materials shows a polynomial curve fit that describes the results (six data only) and, moreover, the intercept of the curve suggests that the minimum Cu level in the Ginna materials for forming Cu-enriched precipitates is 0.046 at. % Cu.

Because materials from the Ginna pressurized water reactor's RPV are of the highest interest due to their higher irradiation dose and demonstrated irradiation-induced embrittlement from the RPV surveillance program, it is encouraging there is sufficient agreement between the two supporting microstructural characterization techniques (i.e., atom probe tomography and small angle neutron scattering) and the hardness measurements (Sokolov et al. 2014).

### *Characterization of Ginna Baffle Former Bolts*

Baffle former bolts removed from some operating LWRs have revealed issues that may be causing cracking. For that reason, baffle former bolts from the Ginna plant may provide valuable material for mechanistic studies on irradiated-assisted stress corrosion cracking and high-fluence effects on swelling and phase transformations. Detailed characterization could provide a more detailed analysis of the irradiated microstructure and help validate high fluence degradation models. In addition, these materials could provide very valuable test material for other studies such as post-irradiation annealing and welding demonstrations. Several highly irradiated baffle former bolts that were removed from Ginna in spring 2011 and stored in the Ginna spent fuel pool have been selected for a pilot study. The following investigations have been proposed to better understand possible cracking mechanisms (Sokolov and Server 2013):

1. Visual examination, photographing surface appearances, and selecting areas of potential cracking
2. Microstructural investigation of areas with potential for cracking
3. Given the baffle former bolt's diameter, three-point bend specimens for fracture toughness and fatigue crack growth rate testing will be machined. The fracture toughness and fatigue crack growth rate tests will be performed at room and operating temperatures (284°C).

### *Summary*

The LWRS Program has been engaged in two key activities that support multiple research tasks: (1) Zion Harvesting Project and (2) Constellation Joint Demonstration Pilot Project. Specific activities within the Constellation Joint Demonstration Pilot Project have included monitoring containment temperature, installation of equipment for monitoring containment rebar and concrete strain, and additional analyses of RPV surveillance coupons from Ginna. Materials harvested or requested from the decommissioned Zion Station Units 1 and 2 include low-voltage cabling, concrete core samples, through-wall-thickness sections of the RPV, and other structures and components of interest to researchers evaluating aging management issues. These collaborative projects provide an invaluable resource of materials for which

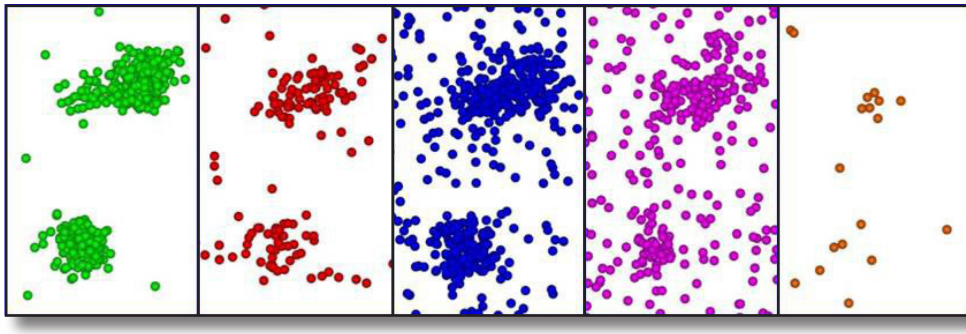

*Figure 11. Atom maps showing Cu-enriched precipitates that include nickel, manganese, and silicon.*

there is limited operational data or experience. Moreover, integrating plant experience with the expert panelists' recommendations, which are documented in the EMDA report, and the Materials Aging and Degradation Pathway's materials research results, lead to these efforts helping inform relicensing decisions and assessments of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior (Busby 2015).

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# Cooperative Research and Development Agreement with Southern Nuclear



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Southern Company Services

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Advanced Instrumentation, Information, and Control Systems Technologies

Southern Nuclear is the nuclear energy operating company for Southern Company, which is one of the largest U.S electric generating companies with over 46,000 megawatts of generation capacity, serving about 4.4 million customers in four southeastern states. Southern Nuclear operates the Farley, Hatch, and Vogtle Nuclear Plants with a combined capacity of 5,318 megawatts. They are also a leader in new nuclear power plant development, with the ongoing construction of Vogtle 3 and 4, two Westinghouse AP-1000 nuclear power plants, which will produce a combined 2,234 megawatts of new nuclear capacity.

Southern Nuclear is one of the original utility participants in the LWRS Program's Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway. In January 2015, Southern Nuclear and the U.S. Department of Energy-sponsored LWRS Program representatives agreed upon a cooperative research and development agreement to focus on a main control room upgrade strategy related to work that was ongoing in the LWRS Program's Hybrid Control Room Pilot Project. Subsequently, Southern Nuclear also developed interest in participating in two additional research areas: computer-based procedures and an advanced outage control center (AOCC).

Discussions occurred with key leaders at the three nuclear sites and the corporate nuclear general office to match up specific-site opportunities with the LWRS Program's pilot project technology development activities. These discussions are summarized in the following paragraphs.

### *Computer-Based Procedures*

The Advanced II&C Systems Technologies Pathway has conducted a series of research and development projects in the area of human performance improvement for nuclear power plant field workers. One such development is in the area of computer-based procedures. A computer-based procedure prototype has been developed that employs a number of efficiency and human performance enhancement features. These features have been derived from basic research and field studies first conducted at Duke Energy's Catawba Nuclear Station and Arizona Public Service's Palo Verde Nuclear Generating Station.

Southern Nuclear is planning to conduct field studies on an enhanced computer-based procedure prototype at their Vogtle Nuclear Plant from May through July 2015. Southern Nuclear will use the prototype on their 92-Day Battery and Battery Charger Inspection and Maintenance Procedure. This quarterly procedure is run on a number of safety-related and non-safety plant batteries and will provide a significant data collection opportunity for technical and human performance aspects of the computer-based procedure technology. Several new performance features have been added to the computer-based procedure, including real-time procedure step authorizations from remote parties, automatic verification of field data measurements, new types of automatically performed computations, automatic completion of procedure data sheets, access to relevant operating experience, automatic access to reference documents, use of "shot clocks" (i.e., clocks used to quicken the pace of an activity) for time-sensitive steps, and seamless transition to related processes, such as the corrective action program initiation form.

### *Business Case for Computer-Based Procedures*

Working with ScottMadden Management Consultants, a business case methodology has been developed by the Advanced II&C Systems Technologies Pathway to quantify the total organizational benefits of a process technology. The purpose of the business case methodology is to provide a structure for building the business case for adopting pilot project technologies in a manner that captures the total organizational benefits that can be derived from the improved work methods. This includes the direct benefit from the proposed pilot project technology to the targeted work process, efficiencies gained in related work processes, and avoided costs through improvement of work quality and reduction of human error.

The Advanced II&C Systems Technologies Pathway and Southern Nuclear have begun to collect data to populate the business case methodology and apply it to the potential business case of computer-based procedures for the Vogtle Nuclear Plant. This activity has the following three main objectives:

1. Provide Southern Nuclear with the basis of an internal business case for considering computer-based procedures as an alternative to today's existing procedure technologies, based on the work efficiencies that can be identified and captured within a nuclear power plant's organization.

2. Provide the broader nuclear industry with a sample business case and demonstrated methodology for use in considering new technologies, such as those being developed through the Advanced II&C Systems Technologies Pathway that can be used as a template for considering similar technologies and their implementations at other nuclear power plants. This will be in the form of an experience report on the business case methodology activity using general information and lessons learned, excluding any vendor or utility proprietary information.
3. Validate the business case methodology through practical application and make continuous improvements to it by obtaining experience and feedback on its use.

### *Nuclear Power Plant Outage Improvement*

Southern Nuclear is leveraging experience from the Advanced II&C Systems Technologies Pathway to develop an AOCC that employs human factors principles and advanced digital technologies in improving coordination, collaboration, and decision making for nuclear power plant outages. This is being coordinated with the Advanced II&C Systems Technologies Pathway's AOCC research and development project, which has the following objectives:

- Provide AOCC capabilities that have already been shown to add value and provide the LWRS Program with additional sources of information that can be used to continue research and development of AOCC technologies and their validation.
- Conduct further research on the potential usefulness of various methods of technology-supported communication and collaboration for outage processes.
- Share lessons learned regarding technology implementation with other utilities engaged in AOCC process improvement efforts.

Initial work has begun with the Farley Nuclear Plant on research and development of advanced collaboration methods for resolving emergent outage issues in a timely and effective manner. A second task will emphasize new methods and technologies for real-time collaboration technologies supported from an AOCC.

### *Control Room Upgrade Strategy*

Southern Nuclear is also participating in research and development of a control room modernization strategy that applies human factors principles to incremental control room improvements. It is part of an Advanced II&C Systems Technologies Pathway's Advanced Hybrid Control Rooms Pilot Project.

The purpose of the collaboration is to develop an incremental approach to control room upgrades that can be executed by a nuclear utility as they conduct instrumentation and control digital upgrade projects to address performance and reliability issues of the legacy systems. The strategy will employ a human-factors based methodology for transforming discrete analog controls to a digital-based system in a manner that enhances operator performance and avoids the introduction of human error traps. It will enable a nuclear power plant owner to more fully exploit the capabilities inherent in digital systems, particularly distributed control systems, to improve operational control in both routine and off-normal plant conditions.

Southern Nuclear's full-scope plant simulator software will be installed in the Human Systems Simulation Laboratory (HSSL), located at Idaho National Laboratory, to provide a high-fidelity working version of the actual control room. An ongoing control room upgrade will be implemented into the simulator software that is operational within HSSL. Workshops will be held in HSSL that involve operators from the selected nuclear power plant. Various scenarios will be selected from normal, abnormal, and emergency operations to compare human factors aspects of the new design with the existing controls and displays.

An assessment of the relevant human factors will be conducted and documented in project reports, including recommendations for optimization. The purpose of these human factors assessments is to establish the existing design constraints and human factors style guide considerations of the existing control room. This will be accomplished by incorporating ongoing control room upgrades into the plant simulator software in HSSL to examine human factors issues related to modification. An operator workshop will be conducted to assess actual operator performance on the basis of the proposed design. Lessons learned and general insights will be used to regularly update existing industry standards, guidance for control room design, and process-related engineering standards.

Southern Nuclear is currently in the process of selecting a suitable control room upgrade project from those at all three of its nuclear power plants to be the subject of the HSSL workshop.

### *Conclusion*

The Southern Nuclear cooperative research and development agreement represents a valuable industry partnership for the U.S. Department of Energy's LWRS Program and its Advanced II&C Systems Technologies Pathway. This agreement brings the experience and knowledge of one of the nation's leading nuclear utilities to bear on the objectives of the LWRS Program. Southern Nuclear's involvement in these projects reflects an innovative approach to pursuing performance improvement through advanced technologies. The insights from this collaboration will be valuable not only for the individual plants that participate in the research, but also for other plants and utilities that are interested in leveraging technology to improve plant performance, reduce human error, and improve plant costs as a way to address the long-term sustainability of their nuclear units.

# RELAP-7 Beta 1.0 Release

**Richard C. Martineau**
**and Curtis L. Smith**

Risk-Informed Safety Margin Characterization Pathway

### *Introduction*

RELAP-7 is the nuclear reactor system safety analysis code currently under development at the Idaho National Laboratory for the RISMC Pathway and is an evolution in the RELAP-series nuclear reactor systems safety analysis codes. RELAP-7 code development takes advantage of the progress made in the past three decades to achieve simultaneous advancement of physical models, numerical methods, coupling of software, multi- parallel computation, and software design. RELAP-7 uses Idaho National Laboratory's open source Multi-Physics Object-Oriented Simulation Environment (MOOSE) framework (www.mooseframework.org) for efficiently and effectively solving computational engineering problems (see Figure 12). Unlike traditional system codes, all physics in RELAP-7 can be solved simultaneously (i.e., fully coupled), resolving important dependencies and significantly reducing spatial and time-based errors relative to traditional approaches. This allows the RELAP-7 development team to focus strictly on systems analysis-type physical modeling (see Figure 13) and gives priority to the retention and extension of RELAP5's system safety analysis capabilities. In addition to the mechanistic calculations representing plant physics, the RELAP-7 design enables it to be integrated into probabilistic evaluations using the RISMC methodology.

> This beta release of RELAP-7 constitutes a significant milestone in the process to develop, validate, and deploy a next generation safety analysis code that will be capable of meeting both current and future industry needs. Continued development and maturation of RELAP-7 will provide advanced analysis capabilities to ensure and enhance nuclear plant safety.
>
> **Stephen M. Hess**
>
> EPRI

The RISMC methodology can be used to optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into safety analysis. The following is a summary of RELAP-7's capabilities that are available in the December 2014 beta 1.0 release of the software.

### *Summary of RELAP-7 Beta 1.0 Release Capabilities*

1. *Analysis Capabilities:*
   a) Incorporates both single and two-phase flow simulation capabilities, encompassing an all-speed, all-fluid (e.g., vapor-liquid, gas, and liquid metal) flow algorithm that is independent of reactor concept (i.e., coolant types). Specific hydrodynamic models include single-phase flow (e.g., liquid water, liquid metal, and single component gas flow), homogeneous equilibrium two-phase flow, and seven-equation two-phase flow with simple closures.
   b) Uses a point kinetics model for simple neutronics analysis.
   c) Provides the capability of performing multi-group radiative diffusion for assembly-homogenized calculations when coupled through the multi-scale radiation transport application referred to as RattleSNake. RattleSNake is a reactor kinetics code that has both diffusion and transport capabilities and is being developed at Idaho National Laboratory based on the MOOSE framework.
   d) Uses one and two-dimensional core heat structures that are strongly coupled with coolant fluids or conjugate heat transfer. Core heat structures are based on fuel-gap-clad substructures.
   e) Incorporates fuels-performance-informed core heat structures through coupling with the BISON fuels performance application. This coupling capability enables fuels performance information to be provided directly to the core heat structures for fuel state evolution (e.g., burnup, depletion, and decay heat), pellet-cladding interaction (e.g., gap closures), thermal property evolution, and clad/centerline fuel temperatures.
   f) Includes a separate Argonne National Laboratory-developed version of RELAP-7 that is focused on sodium fast reactors, sodium coolant, and fast spectrum point kinetics.

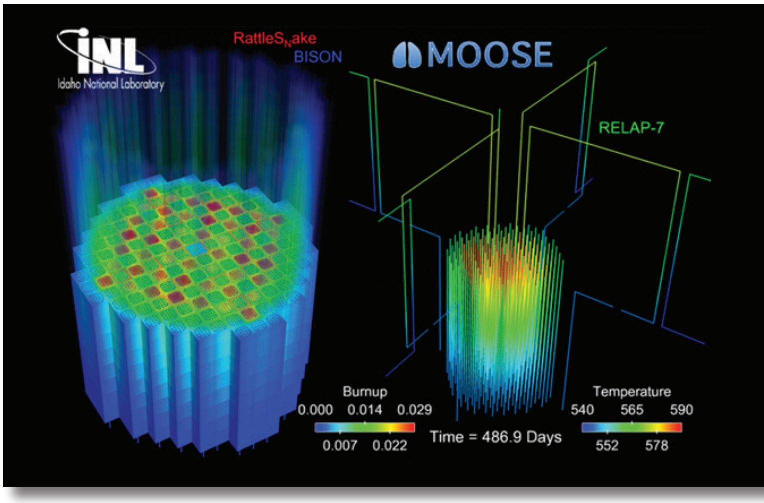



***Figure 12. Coupled MOOSE and RELAP-7 detailed core calculation.***

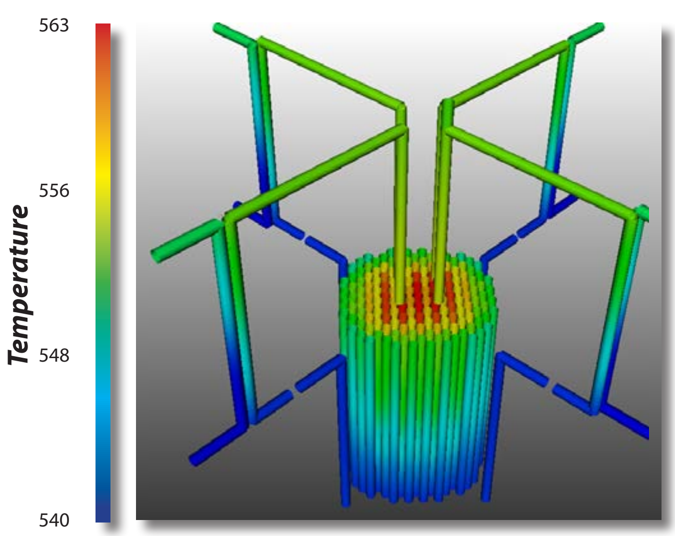
g) Employs various component modules, including pipes, core channels, heat exchangers, branches, valves, and pumps.

2. *Range of Applicability:*

a) Single-phase, steady-state and transient analysis for pressurized water reactor applications (note that the steam generator and pressurizer models for a pressurized water reactor are still under development and not available yet), using a homogeneous equilibrium two-phase flow model.

b) Boiling water reactor steady-state and transient analysis (including station blackout) using a homogeneous equilibrium two-phase flow model.

***Figure 13. Systems Analysis Evaluation using RELAP-7.***




c) Boiling water reactor steady-state and transient analysis (including station blackout) using the seven-equation, two-phase flow model for the reactor core and homogeneous equilibrium two-phase flow model for the balance of the plant.

3. *Equation of State/Properties:*

a) Ideal gas equation of state for single-component gas (e.g., helium, nitrogen, etc.).

b) Stiffened gas equation of state for single-phase liquid water and simplified water steam mixtures.

c) Linearized equation of state for general applications such as single-phase water and sodium.

4. *Target User Community*

Currently, the RELAP-7 code is still undergoing development, optimization, and verification and validation. The primary objective of the December 2014 beta release is to obtain feedback and suggestions for improvement on usability and applicability from the user community. Therefore, this beta release will be limited to a selected set of users who are experienced in developing and using reactor systems safety analysis codes such as RELAP5, TRAC, and TRACE. In addition to the software, supporting documentation is available including a RELAP-7 Theory Manual and a RELAP-7 Users Guide

The RELAP-7 Theory Manual and User's Guide can be found at https://relap7.inl.gov or www.inl.gov/lwrs.

# Recent LWRS Program Reports

## *Technical Integration Office*

- ***DOE-NE Light Water Reactor Sustainability Program and EPRI Long-Term Operations Program – Joint Research and Development Plan, Revision 4, INL-EXT-12-24562, April 2015.***
  https://lwrs.inl.gov/Technical%20Integration%20Office/INL-EXT-12-24562_LWRS-LTO_Joint_RD_Plan_Rev_4.pdf

## *Materials Aging and Degradation*

- ***Assessment of Cable Aging Equipment, Status of Acquired Materials, and Experimental Matrix at the Pacific Northwest National Laboratory***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/M2LW-15OR0404012.PNNL-24198.Fifield.pdf
- ***Thermal aging modeling and validation on the Mo containing Fe-Cr-Ni alloys***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS_Semiannual%20report_2015_thermodynamics_final.pdf
- ***System-Level Heat Transfer Analysis, Thermal-Mechanical Cyclic Stress Analysis, and Environmental Fatigue Modeling of a Two-Loop Pressurized Water Reactor: A Preliminary Study***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ANL-LWRS-15-01.pdf
- ***Reestablishing Capability at PNNL and Aging and Testing Schedule***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/CASS_Aging_TestMatrix.pdf
- ***Scoping/Design Study on Optimum Configuration for Combined Thermal/Radiation Aging of Cable Insulation Samples at ORNL***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS.M3LW-15OR0404015.Duckworth.pdf
- ***Thick Concrete Specimen Construction, Testing, and Preliminary Analysis***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ORNL-TM-2015-72_Final.pdf
- ***Assessment of the Quality Program for the Light Water Reactor Sustainability Program's Materials Aging and Degradation Pathway at the Oak Ridge National Laboratory and the Pacific Northwest National Laboratory***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS_MAaD_Assessment_Report2015.pdf
- ***Assessment of Additional Key Indicators of Aging Cables in Nuclear Power Plants – Interim Status for FY 2015***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/M3LW-15OO404022-PNNL-24309.pdf

## *Risk-Informed Safety Margin Characterization*

- ***RAVEN User Manual***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/INLEXT-15-34123_raven_user_manual_Rev_1.pdf
- ***Reactor Pressure Vessel Fracture Analysis Capabilities in Grizzly***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/grizzly_fracture_milestone.pdf
- ***Light Water Reactor Sustainability Program Seismic Data Gathering and Validation***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/INL_EXT-15-34425.pdf
- ***Simulation and Non-Simulation Based Human Reliability Analysis Approaches***
  https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/INL-EXT-14-33903-Simulation_and_Non-Simulation_Based_Human_Reliability_Analysis_Approaches.pdf

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- ***Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis***
  https://lwrs.inl.gov/Reactor%20Safety%20Technologies/ANL_NE-15_4%20gap%20final%203-31-15.pdf
- ***Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk Phase 1 Study, 3002004449, Electric Power Research Institute, Technical Update, November 2014.***
  https://lwrs.inl.gov/Reactor%20Safety%20Technologies/Crosswalk-phase1-final.pdf

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