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29    Uticom Systems, Inc.                    Jackie Waller jackie@uticom.net
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Directory Listings on page 41.

Nuclear Plant Journal Rapid Response Fax Form

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May-June 2014
SETTING THE STANDARD IN OUTAGE PERFORMANCE

With the cooperation and close support of our clients, CB&I’s efficient and safe execution during facility refueling outages has helped break industry records and define best practices. Our leading U.S. nuclear maintenance services, integrated project planning and execution are led by expert teams of mobile professionals providing superior performance to our fleet and site alliances.

Contact CB&I for a full range of services in nuclear design, EPC, modifications and maintenance for your nuclear power project.

EXTENDED POWER UPRATES
OUTAGE PLANNING AND EXECUTION
PLANT MODIFICATIONS AND MAINTENANCE
DECONTAMINATION AND DECOMMISSIONING
Joining Forces

ITT Goulds Pumps and HydroAire have joined forces—creating a powerhouse for pump reliability.

HydroAire is now the exclusive licensee for manufacturing ITT Goulds pumps and replacement parts, providing the nuclear power industry with superior ASME code and safety related pumps, parts and aftermarket services.

For over 50 years ITT Goulds Pumps has provided nuclear pumps for critical applications such as component cooling water, safety related service water, boric acid transfer, spent fuel pool cooling, containment spray and safety related fire systems. Over the past 20 years HydroAire has methodically built the industry’s strongest independent nuclear pump services organization with a reputation for improving pump performance through unequaled customer service centered around responsiveness to our customer’s needs.

Our long successful history developed each of our legacies to become your trusted partner to support not only current products but also obsolete and orphaned pumps with nuclear qualified replacement pumps, parts and service on your site’s most critical equipment.
Our Comprehensive Services

OEM Goulds Pumps, Parts & Accessories  l  Nuclear Pump Repairs
Field Service Support for Installation & Start-up  l  Certified Performance Testing
Pump System Reliability Engineering Services  l  Customer Training

To learn more about Hydro’s comprehensive services, please contact us at 800-223-7867 or visit us at hydroinc.com
For information on Goulds products, parts or services contact us at GouldsProducts@hydro-aire.com.
New Energy

Olkiluoto

The testing of the Instrumentation & Control (I&C) system for the Olkiluoto 3 EPR reactor began on April 1, 2014 at AREVA’s site in Erlangen, Germany. The I&C system, used to monitor and control the nuclear power plant, is a key component in reactor operations.

Based on the quality of the I&C documentation submitted by the AREVA Siemens consortium, TVO agreed to begin the operation.

Philippe Knoche, chief operating officer of AREVA, said: “The Olkiluoto 3 project has reached an important step. Our technology meets the most stringent safety criteria and has already been licensed in 16 countries to equip nearly 80 nuclear reactors with 14 different designs. This technology has been certified by the French, British, and Chinese safety authorities for the EPR reactor. Through our technological leadership, we are able to advance in Finland.”

Following these testing operations, the I&C cabinets will be transported to Finland and installed at the Olkiluoto site in order to begin the testing phase before the reactor commissioning.

Contact: Julien Duperray, telephone: 33 1 34 96 12 15, email: press@areva.com

Finland

Fennovoima’s owners Voimaosakeyhtiö SF and RAOS Voima Oy have made the binding decision to construct and finance Fennovoima’s nuclear power plant in Pyhäjärvi.

Fennovoima supports Finland’s competitiveness and the operations of its owner companies. The project increases Finland’s self-sufficiency in electricity generation and strengthens the security of supply. According to the agreed schedule, the power plant will generate electricity for its owners at cost price starting from 2024. The power plant will employ directly thousands of people during construction period and generates durable demand for various services in Finland and especially in Northern Ostrobothnia.

The Fennovoima general meeting that was held today has nominated a new board. The Chairman of the Board is Juhani Pitkäkoski, Senior Vice President of M&A of Caverion Corporation and the vice chairman is Anastasia Zoteeva, Deputy Director General for Business Development of Rusatom Overseas. The Members of the Board are Aimo Eloholma, former Board Chairman of Megafon (Russia); Esa Lager, former CFO of Outokumpu Oyj; Juha Mäkitalo, Attorney-at-Law; Stefan Storholm, CEO of Katttermo Group and Djurica Tankosic, President of Global Nuclear of Worley Parsons. The Deputy Members of the Board are Antti Koskelainen, Energy Director of Outokumpu Oyj and Jussi Lehto, CEO of Keravan Energia.

The Ministry of Employment and the Economy of Finland is preparing a proposal for Fennovoima’s application to supplement the Decision-in-Principle. If the Government approves the supplement, the application will proceed to Parliamentary ratification.


Sanmen

Westinghouse Electric Company announced the successful completion of a significant milestone at the AP1000® nuclear power plant currently under construction in Sanmen, China, when the Unit 1 main control room (MCR) was declared operational by Sanmen Nuclear Power Company, Ltd. (SMNPC) and State Nuclear Power Technology Corporation (SNPTC). Once the plant is online and operating, certified reactor operators will work from the MCR to monitor and control plant processes.

In order to achieve the MCR Operational milestone, several key instrumentation and control (I&C) systems, including plant control system (PLS), data and display system (DDS) and the operations and control system (OCS), were integrated and tested to ensure key plant functions could be monitored and controlled from the MCR. Once these systems were integrated, the AP1000™ plant’s digital control system will complete multiple commissioning activities, including the commissioning of the 220 volt and 10 kilovolt electrical systems, service and instrument air, and demineralized water systems.

Westinghouse delivered all of the I&C systems, as well as the design documentation, engineering services and mechanical equipment in support of this milestone.

Achieving the Main Control Room Operational milestone is a major accomplishment for Westinghouse, SMNPC and SNPTC. David Howell, senior vice president, Westinghouse Automation and Field Services. The integration of key I&C systems and the operability of the main control room will now facilitate other key testing activities at the world’s first AP1000™ nuclear power plant.

Contact: Sheila Holt, telephone: (412) 374-6379, email: holtsa@westinghouse.com

Shidaowan

The pouring of concrete for the basemat of the first HTR-PM unit-a demonstration high-temperature gas-cooled reactor - at Shidaowan in China’s Shandong province was recently completed. Another 19 of the small modular reactors could follow.

Plant constructor China Nuclear Engineering Corporation (CNEC) announced that the pouring of concrete to complete the basemat for the HTR-PM was finished on March 29, 2014. Construction of the reactor building itself will now begin.

The demonstration plant’s twin HTR-PM units will drive a single 210 MWe turbine. It is expected to begin operating around 2017. Eighteen further units are proposed for the Shidaowan site, near Rongcheng in Weihai city.

Work began on two demonstration HTR-PM units at China Huaneng Group’s Shidaowan site in December 2012. Since then, the foundations and columns to support the reactor building
have been installed. China Huaneng is the lead organization in the consortium to build the demonstration units with CNEC and Tsinghua University’s Institute of Nuclear and New Energy Technology (INET), which is the research and development leader. Chinergy, a joint venture of Tsinghua and CNEC, is the main contractor for the nuclear island.

Two demonstration CAP1400 units - scaled up versions of Westinghouse’s AP1000 - are also planned for the Shidaowan site. About 80% of components for these will be made in China. The Shidaowan site is part of the larger Rongcheng Nuclear Power Industrial Park.

Contact: World Nuclear News, website: www.world-nuclear-news.org

Turkey Point
Approval for two new reactors at Turkey Point and the required transmission lines has been granted by Florida state authorities.

State governor Rick Scott and a majority of cabinet members voted in favour of the new nuclear project yesterday, approving at the same time a route for new transmission lines that had been contested by some city officials.

This brought to a conclusion the state’s ‘one-stop-shop’ approval process for new power plants that included an eight-week hearing last year after five years of liaison with community leaders.

Florida Power and Light’s (FPL) plan is to add two AP1000 units to Turkey Point nuclear power plant, which already features two reactors built in the 1970s and is part of complex including two gas-fired units and an oil-fired unit.

FPL has been officially planning to expand nuclear generation at Turkey Point since 2006, when it informed the US Nuclear Regulatory Commission (NRC) that it intended to file for a construction and operating license for new units. It made the application in 2009, and in 2011 the Florida Public Service Commission allowed FPL to add a small charge to its customers’ bills to fund the nuclear investment.

The license application for the new reactors remains with the NRC, which could complete its work by early 2016. FPL hopes to secure these permissions and build the plant in time to generate electricity from the first new reactor in mid-2022.

Contact: World Nuclear News, website: www.world-nuclear-news.org

Xudabao
The Xudabao site in China’s Liaoning province has been approved for the construction of the first two of six AP1000 units planned there.

The National Nuclear Safety Administration (NNSA) announced that it has approved the site selection for Xudabao units 1 and 2. It said that the site would meet site-related aspects of nuclear safety regulations.

The site of the proposed Xudabao nuclear power plant is in Xingcheng City on the island of Hulu, in the northeast of the coastal province of Liaoning. While the initial phase of the project will comprise two AP1000s, a further four such units are planned for the site.

Site preparation at Xudabao began in November 2010. The National Development and Reform Commission gave its approval for the project in January 2011. However, following the Fukushima accident two months later, Chinese authorities suspended the approval process for new plants. The Xudabao plant still requires a construction license from the State Council.

In October 2012 China announced that approvals for inland plants would be delayed until after 2015. Parts that had been intended for the Taohuajiang plant in Hunan province - where four AP1000 units are planned - were subsequently earmarked for Xudabao. Manufacture of the steel containment for the first two Xudabao units was launched in July 2013 by Shandong Nuclear Power Equipment Manufacturing Co Ltd.

In September 2007, Westinghouse and its partners the Shaw Group received authorization to construct four AP1000 units in China: two at Sanmen in Zhejiang province and two more at Haiyang in Shandong province.

Contact: World Nuclear News, website: www.world-nuclear-news.org
Utility, Industry & Corporation

Utility

Nuclear Fleet

Three nuclear energy plants owned by Constellation Energy Nuclear Group, LLC officially joined Exelon Generation’s fleet of nuclear plants, expanding what was already the nation’s largest commercial nuclear operation.

The three CENG plants include five reactors capable of generating more than 4,200 megawatts at full power. The three plants are R.E. Ginna Nuclear Power Plant in Ontario, New York, Nine Mile Point Nuclear Station in Scriba, New York, and Calvert Cliffs Nuclear Power Plant in Lusby.

The NRC approved license transfers for the three plants on Tuesday, March 25, 2014.

The consolidation, which followed the March 2012 merger between Exelon and Constellation Energy Group Inc., expands the fleet of reactors Exelon operates to 23 nuclear generating.

Michael J. Pacilio, Exelon Nuclear’s President and Chief Nuclear Officer, will lead the newly expanded fleet, which will operate under Exelon Nuclear’s management model, a driver of success and consistent operations at all stations. Maria G. Korsnick will remain chief nuclear officer of CENG.

Contact: telephone: (630) 657-3602.

Top Industry Practice

A team of Southern Nuclear engineers at the Edwin I. Hatch Nuclear Plant were recently honored at Nuclear Energy Institute’s Nuclear Energy Assembly in Phoenix, Arizona. They received the nuclear industry’s highest honor – a TIP Award – for developing a Top Industry Practice. The team claimed the 2014 GE Hitachi Vendor Award for its participation in GE Hitachi’s development of the Stinger™ Automated IVVI, a new first-of-a-kind tool for performing in-vessel visual inspections.

The radiation-tolerant Stinger™ tool features a unique extendable inspection camera and weld cleaning system that’s operated by workers positioned hundreds of feet away from the vessel cavity.

The inspection camera leader system is maneuverable. It can pan, tilt and rotate in five axes of motion, even around corners, to obtain the optimal view. For the Unit 2 outage, inspection coverage increased 20-30 percent, and the visual quality of the exams also improved.

With traditional in-vessel visual inspections, several outage activities must be suspended during fuel shuffles and new fuel replacement, to protect workers. With ROV technology, which allowed inspection operators to work safely from a remote location, employees’ exposure to radiation showed a marked improvement.

Contact: telephone: (205) 992-5395.

Industry

Safety Enhancement

On April 8, 2014, the Nuclear Energy Agency (NEA) in co-operation with the Nuclear Regulation Authority (NRA) of Japan held an international conference on global nuclear safety enhancement in Tokyo, Japan. High-level experts from nuclear regulatory authorities in France, Japan, Korea, Russia and the United States reviewed international developments in nuclear safety since the Fukushima Daiichi nuclear power plant accident in March 2011, as the international community works to strengthen nuclear safety at the global level. At the conference, OECD Secretary-General Angel Gurría spoke about sound energy policies for economic and social development and insisted on very high levels of safety as the first condition for using nuclear power. NRA Chairman, Dr. Shunichi Tanaka also stressed the importance of the independence, technical capability and transparency of the regulatory authorities, as well as a strong safety culture.

Contact: Cynthia Gannon-Picot, telephone: (205) 992-5395, email: press@oecd-nea.org.

Leveraging Technology

Standard outage management communications tend to be conducted over the phone, in face-to-face reports and in outage center briefings. Idaho National Laboratory observed Arizona Public Service’s spring 2013 outage control center technologies for the next outage in the fall of 2013, which involved the challenge of repairing a reactor bottom vessel-mounted instrumentation leak.

The entire organization mobilized for the repair, supported by collaborative technology. They used Microsoft OneNote to collect, organize, and share information, including photos, drawings, schedules, and action items. Every aspect of inspections, repairs and recovery planning was in one easily searchable, consistently organized place, network-accessible from any work station on site.

These communication technologies ensured that the outage and its repair, which earned Nuclear Energy Institute’s TIP Award for Maintenance and the Westinghouse Combustion Engineering Vendor Award, were completed on schedule. The team helped to save 40 critical path days compared to the previous repair of its kind, for a direct savings cost of $48 million.

INL plans to publish a report documenting implementation of advanced outage control center technologies, which are transferable across the industry.

Contact: Shawn St. Germain, telephone: (208) 526-9575, email: shawn.stgermain@inl.gov.

Director-General

The OECD Secretary-General announced the appointment of Mr. William D. Magwood as the next Director-General of the OECD Nuclear Energy Agency (NEA). He will be succeeded by Luis E. Echávarri who is

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retiring at the end of April 2014 after nearly 17 years at the head of the NEA. Mr. Magwood – who is currently one of the five Commissioners of the US Nuclear Regulatory Commission, was previously Director of Nuclear Energy at the US Department of Energy and has served as the Chair of the NEA Steering Committee – will take up his duties on 1 September, 2014.


Safety Scholarships

The World Association of Nuclear Operators (WANO) marked the 25th year of its creation recently by establishing £2,500 ($2,518) per student scholarships for four students aged 25 and under who are pursuing post-secondary (college) degrees in nuclear power generation.

Interested students will be asked to submit their biography and a 250-word essay on the importance of making safety their top priority while pursuing a career in nuclear energy. An online submission form will be added to www.wano.info in September 2014 and essays will be accepted until Dec. 31, 2014. All submissions will be judged by members of WANO’s Executive Leadership Team.

In addition to the scholarships, the four winners will also be WANO’s special guests at its 2015 Biennial General Meeting in Toronto, where the world’s top nuclear executives will gather to reaffirm their commitment to WANO and its mission.

Contact: Steve Cannon, telephone: 44 20 7495 9242, email: steve.cannon@wano.org.

I&C Installation

AREVA Inc. successfully installed its TELEPERM® XS Platform digital Instrumentation and Control (I&C) system at a third nuclear reactor in the United States.

Incorporating lessons learned from the previous two installations, the AREVA team successfully supported the installation and testing of the third system ahead of schedule with no safety issues.

AREVA’s TELEPERM® XS I&C system is a fully digital system, used to monitor and control the safety of a nuclear plant.

Contact: Dolline Hatchett, telephone: (704) 517-9837, email: dolline.hatchett@areva.com.

NRC License

AREVA TN has received a U.S. Nuclear Regulatory Commission (NRC) license for the NUHOMS® MP197HB Transport Package. The MP197HB is the only transport package that meets the rigorous NRC requirements to safely transport high burnup used nuclear fuel and has the highest heat-load capacity in the industry of up to 32 kilowatts.

The MP197HB is also the first universal transport package that is licensed for nine different canister types from various suppliers currently in use at utility sites, including AREVA TN’s NUHOMS® high-performance dry shielded canisters and the AREVA TN NUHOMS Radwaste Canister. The NUHOMS® MP197HB transport package is approved for use by rail, truck or marine transport, and will be available as early as 2015.

Contact: Mary Beth Ginder, telephone: (301) 841-1703, email: Marybeth.ginder@areva.com.

ASME Certified

Doosan Vina (Doosan Heavy Industries Vietnam Co., Ltd located in Quang Ngai Province, Vietnam) released a statement that they are certified by ASME as a supplier of nuclear products.

The certification process took more than a year to complete and culminated in December 2013 with a visit to Vietnam by a team of seven ASME nuclear specialists from the USA, Korea and Japan. They came to Doosan Vina’s 110 hectare industrial complex in the Dung Quat Economic Zone of Quang Ngai Province to survey the results of the ASME team’s year long review of operations. Doosan Vina is the first company in South-East Asia to successfully complete the ASME nuclear product fabrication certification process.

Doosan Vina was awarded six new ASME Nuclear certificates, (NPT, NA, NS, Site NPT, Site NA, Site NS) after the 2013 nuclear survey. The company also received four ASME non-nuclear certificates, (U, U2, PP, S) which were initially awarded by ASME in 2008.

Contact: website: www.doosan.com/doosanvina.

“N” Stamp

Flowserve Corp., a provider of flow control products and services for the global (Continued on page 14)
infrastructure markets, has announced that its UK ball valve manufacturing operation has been awarded ASME III “N” stamp approval.

The certification covers all ASME III valve classifications for the nuclear power industry including Class 1, which allows Flowserve Flow Control UK through its manufacturing facility in Burgess Hill, West Sussex to supply Worcester Controls ball valves to the most critical service applications on the nuclear island. Section III of the ASME Boiler and Pressure Vessel Code is built around quality standards that enhance safety throughout a nuclear site and is the most widely used nuclear pressure vessel code around the world.

Flowserve Flow Control UK is the only UK manufacturer of ball valves to be awarded the ASME III “N” stamp. It has also been approved as a Material Organization for the supply of ferrous and non-ferrous materials.

Contact: Lars Rosene, telephone: (972) 443-6644.

Technical Review
The NRC has completed its technical review of the design of GE Hitachi Nuclear Energy’s Economic Simplified Boiling Water Reactor (ESBWR) and last week the commission released a rulemaking schedule as part of a process that is expected to lead to certification of the ESBWR later this year.

GEH CEO Caroline Reda said, “We are excited to reach this important milestone in the development of the ESBWR. We look forward to an expeditious rulemaking process to certify the ESBWR design, paving the way for the reactor’s construction in the U.S. and supporting opportunities for us globally.”

Contact: Jon Allen, telephone: (910) 819-2581, email: jonathan.allen1@ge.com.

Vertical Lifter
Konecranes, a global manufacturer of lifting technology, offers the dynamic ATL Vertical Lifter to meet the handling requirements of a variety of industries.

Boasting a lifting capacity of up to 1,600 kg and absorption of torque up to 4,000 Nm, the ATL Vertical Lifter from Konecranes offers diverse, high-quality drive concepts with pneumatic balance controls or electrical chain hoists.

Konecranes’ pneumatic manipulators are ceiling-based solutions made up of an XA crane system and a telescopic lifting device combined with a customized gripping unit. Their modular design allows flexibility and adaptability to meet the particular needs of individual operations.

Unlike chain or rope hoists, the ATL Vertical Lifter from Konecranes can lift loads outside the center of gravity and its balancing mode makes the handling of parts feel nearly weightless. High-quality telescope guides enable precise, swaying lifting and vertical manipulating.

Contact: Todd Blair, telephone: (937) 525-5560, email: todd.blair@konecranes.com.

Fuel Assembly Mapping
Outage crews on Unit 1 at the Byron Generating Station were able initiate the upper internals lift earlier than scheduled, based in part on the rapid completion of fuel assembly mapping by engineers from Newton Labs using the Newton NM200E Core Verification System. Only 49 minutes were required to fully map all 193 assemblies, finishing well within the five and a half hours scheduled for this activity on the Critical Path. The total deployment-to-recovery elapsed time for the NM200E system was two and a half hours.

The NM200E tool has been adopted by Exelon as their primary method for determining nozzle position in their PWRs and has been deployed at a number of other nuclear plants in the U.S. and France. The Newton-developed NM200E software determines the positions of Westinghouse and AREVA PWR fuel assemblies based in part upon S-hole positions, and it is also able to map the more symmetrical Babcock and Wilcox nozzles.

Contact: Eric Yates, telephone: (425) 251-9600, email: eyates@newtonlabs.com.

New Location
Nuvia Group, a subsidiary of Soltanche Freyssinet, announced that Nuvia-USA has located an office in Charlotte, North Carolina in January 2014. Nuvia-USA is a specialty service and product provider in the nuclear industry meeting today’s challenges in Fire Protection, Plant Life Extension, New Build, Decommissioning/Dismantling and Waste Management operations, and Nuclear Measurement equipment which include detectors, digital analyzers and waste management systems. To better serve their customers Nuvia-USA is in the process of qualifying the latest technology in fire dampers; fire and flood penetration protection and fire barrier penetration products; while meeting the demands in supporting ongoing decommissioning and dismantling operations with its specialty diamond and laser cutting tools for use in normal and extreme hostile environments, which are in use and being applied to reduce and manage RISK during these operations. Nuvia-USA has been selected to be the only authorized distribution of Mecatiss passive fire barrier products which also provides training and reviews for new or existing installations; as well as providing replacement products for the extensive product line during normal maintenance, and re-application to existing installations in the United States, where there are six nuclear units that has UL Listed Mecatiss installations.

Contact: Jacques Cardon, telephone: (704) 998-5510, email: jacques.cardon@nuvia-usa.com.

Nuclear Agreement
Westinghouse Electric Company and Ontario Power Generation, Inc. (OPG), through its subsidiary, Canadian Nuclear Partners (CNP), today announced an agreement that would enable the two companies to work together on a wide range of global nuclear projects.

Under the agreement, the companies will consider a diversity of nuclear projects including refurbishment, maintenance and outage services, decommissioning and remediation of existing nuclear facilities, and new nuclear power plants.

Westinghouse and OPG have collaborated recently on several key projects. Currently, Westinghouse is performing work to design filtered containment vents at OPG’s Darlington Nuclear Power Station as part of the site’s refurbishment program.

Contact: Sheila Holt, telephone: (412) 374-6379, email: holtsa@westinghouse.com.
Siempelkamp supplies equipment to open and close reactor pressure vessels and primary circuit components. As a specialist for the design, construction, delivery and maintenance we offer purpose built solutions including:

- multiple stud tensioners
- single stud tensioners
- turning and transportation devices
- accessories
- refurbishment to existing equipment
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New Products

Weld Inspection

AREVA has developed a cutting-edge tool for nuclear reactor vessel inspections that optimizes both the safety and reliability of operations for nuclear facilities. The automated tool, known as the Lower Girth Weld Inspection Tool (LGWIT), allows plant personnel to perform inspections remotely through narrow gaps in the reactor vessel internals.

The tool’s remote operation replaces the need for direct human intervention. This is both more efficient and contributes to significant cost savings for each inspection performed. The inspection of reactor vessel internals is essential to ensure the safe, long-term operation of nuclear power plants. This is also a key requirement in the license renewal process.

AREVA designed this tool to meet the criteria of the Electric Power Research Institute’s Materials Reliability Program and the U.S. Nuclear Regulatory Commission’s NUREG 1766 (Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2) regulation. The successfully proven LGWIT tool is currently in use by AREVA at outages for two major U.S. utilities.

Contact: Dolline Hatchett, telephone: (704) 517-9837, email: dolline.hatchett@areva.com.

New Technology Meters

OTEK Corporation has released a new series of meters - The NTM (New Technology Meters). This new series changes the way we look at digital bargraph meters and introduces an exclusive input failure alert and alarm system.

The new series of meters offers operator friendly digital display as well as an automatic tricolor bargraph with color set point. There are no moving parts in the design, making it durable and reliable. High quality products allows OTEK to offer an industry unique lifetime warranty.

A featured model in the series is the NTM-3. This meter is ANSI 4” switchboard replacement meter. The NTM-3 is available in plastic or metal housing and meets military or nuclear specifications. You can build up to three channels to house in one meter. Additionally, the meters can be loop, signal or externally powered.

You can have the visibility and accuracy of a digital meter and the low power consumption of an analog meter all in one. There are many more options available in the series.

Contact: telephone: (520) 748-7900, email: sales@otekcorp.com.

Valve Actuator

Rotork-Hiller has announced a pneumatic valve actuator design breakthrough that greatly reduces space requirements while meeting specifications and regulatory requirements for critical failsafe applications in the nuclear power industry.

The new design incorporates the actuator, nitrogen storage accumulator, and all associated controls within a compact, standalone package to provide springless, totally pneumatic operation. One example for a 1-1/2” stroke linear actuator used in a plant fire control system has envelope dimensions of only 18 x 23 x 27 inches.

Rotork-Hiller engineers can configure the new actuator design to meet customer-specific needs.

Contact: Mike heuseveldt, telephone: (585) 247-2304, email: info@rotork.com.

Field Machining

POWERMAQ has launched its first global field services company. From its U.S. headquarters in Portland, Ore., the new venture will offer expert machining and repair services for heavy industry worldwide.

POWERMAQ USA is staffed with professional managers, engineers, technicians, and field machining experts. Several of the technicians and machinists have more than 20 years’ experience in on-site machining. Their deep experience includes projects in nuclear facilities, oil and gas refineries, mining operations, as well as subsea environments and a host of other industrial locations.

To perform the difficult repairs required, the POWERMAQ USA crew is equipped with state-of-the-art portable machine tools which are known for peak performance in challenging on-site repairs -- whether refacing a severely corroded flange in a refinery, repairing leaks to pipelines, offshore or undersea, or doing turbine maintenance at a power plant.
POWERMAQ USA technicians can mobilize within 24 hours to adeptly handle emergency repairs, and also are an excellent supplement to a company’s existing workforce during a planned shutdown or outage.

Powermaq’s machinists and technicians have extensive experience making repairs in nuclear plants, in fact, it is one of their special areas of expertise.

Contact: Philip Bauder, telephone: (503) 523-1581, email: bauder@powermaq.com.

Contracts

Outage & Maintenance

PSEG Nuclear has extended through 2022, the contract signed with AREVA for outage and maintenance activities at the Salem units 1 and 2, and Hope Creek nuclear generating stations in New Jersey.

Following the first contract signed in July 2013, AREVA will continue to provide its technical support to PSEG’s pressurized and boiling water reactors through refueling, inspections and steam generator services. This contract reinforces AREVA’s position as a service supplier in the boiling water reactor market.

Contact: Julien Duperray, telephone: 33 1 34 96 12 15, email: press@areva.com.

Stators Replacement

AREVA has been awarded a contract by Korea Hydro & Nuclear Power (KHNPC), subsidiary of the Korea Electric Power Corporation (KEPCO) for the supply and the replacement of six stators for the Kori Nuclear Power Plant Units 3 and 4.

These components ensure the reliability of the reactor coolant pump which is a fundamental element for the safe and efficient operation of the reactor’s primary circuit.

AREVA will design and manufacture the six stators in the facility located at Jeumont, in Northern France. The delivery and replacement operations are expected to begin in June 2015 and finish in 2018.

This contract is part of the “Forward Alliance” program, which offers products and solutions developed by AREVA to support utilities in the optimization of the long term safety of their plants.

Contact: Julien Duperray, telephone: 33 1 34 96 12 15, email: press@areva.com.

Fuel Deliveries

Westinghouse Electric Company and the National Nuclear Energy Generating Company of Ukraine (NNEGC Energoatom) agreed to a contract extension for fuel deliveries to Ukrainian nuclear power plants through 2020. The contract provides for the continuation of the long-standing partnership between the two companies in providing competitive and secure nuclear fuel supplies for Ukraine’s reactor fleet.

Westinghouse originally signed a contract for nuclear fuel in 2008. The contract amendment will extend Westinghouse deliveries through 2020. Under terms of the contract executed with Westinghouse Electric Sweden AB, Westinghouse will produce the fuel at its fabrication facility in Västerås, Sweden. The Vasteras facility is one of the largest and most modern fuel plants in the world, serving every major market in Europe and South Africa.

Contact: Sheila Holt, telephone: (412) 374-6379, email: holtsa@westinghouse.com.

MOHR Test and Measurement LLC (MOHR) is supplying its EFP-IL SFPI system for remote monitoring of spent fuel pool water level to at least 20 U.S. nuclear plants including Utilities Service Alliance member stations; Columbia, Cook, Cooper, Fermi 2, Fort Calhoun, Hope Creek, Monticello, Prairie Island, Salem, and Susquehanna. The EFP-IL uses ultra-wideband guided radar technology that lets the system accurately measure liquid level through more than 1000 ft. (305 m) of coaxial cable, characterize frothing/boiling environments, and perform real-time in-situ automatic calibration. The system includes integrated 7 day battery backup and meets all NRC Order EA-12-051 (To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation) requirements including seismic qualification to IEEE 344.

Contact: Brandt Mohr, telephone: (888) 852-0408, email: bcmohr@mohrandassociates.com.

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accident condition from basic concepts to describes the fuel behavior of each loss-of-coolant accidents. This report (1) reactivity-initiated accidents and (2) following two types of accident conditions:


This NUREG report considers the following two types of accident conditions: (1) reactivity-initiated accidents and (2) loss-of-coolant accidents. This report describes the fuel behavior of each accident condition from basic concepts to the current state of the art.


The purpose of this knowledge management NUREG and DVD is to preserve the history and impact of the fire at the Browns Ferry Nuclear Plant (BFN) on regulations and to educate future generations of safety professionals. This is the second report in the NUREG/KM-Series.


This NUREG presents research conducted across a wide variety of disciplines, ranging from fuel behavior under accident conditions to seismology to health physics. This research provides the technical bases for regulatory decisions and confirms licensee analyses.


This digest provides fire related information for three major subject areas: current nuclear power plants, next generation nuclear power plants, and non-reactor and nuclear materials.


This knowledge management digest and accompanying DVD contain historical documents relating to the technical and regulatory aspects of the reactor pressure vessel head incidents at Davis-Besse Nuclear Power Station in 2002 and 2010.


The 2013 New Reactor Program annual review highlights the diverse array of activities undertaken and showcases the resilience and dedication of the employees to manage a myriad of projects toward safe closure.

The above NUREG-series publications and other NRC records can be electronically accessed at NRC’s Public Electronic Reading room at www.nrc.gov/reading-rm.html

EPRI


Cooling water in leakage can result in huge costs for power plants as a result of plant derates and outages to locate and mitigate leaks, vendor services to help with leak location detection, or extended plant outages necessitated to address catastrophic inleakage. Based on current studies and on recommendations made by the Institute of Nuclear Power Operations (INPO), new leak detection targets are being considered with leak flow rates as low as 1–2 gallons (3.8–7.6 liters) per day.


This paper is the first of a two-part series that examines advanced welding applications and issues for their use in the nuclear power industry. It focuses on primary nuclear welding applications and advanced welding processes/applications that can address industry weldability issues.


This report, in conjunction with EPRI report 3002000494, Nuclear Maintenance Applications Center: Bolting Performance Demonstration Unit (BPU) Drawings and Instructions, provides the information needed to construct a BPU with an updated digital output and display so that the BPU can be used by power plant personnel for training on correctly assembling bolted joints.

The above EPRI documents may be ordered by contacting the Order Center at (800) 313-3774, Option 2, or email at orders@epri.com.
1. 22nd International Conference on Nuclear Engineering (ICONE 22), July 7-11, 2014, Prague, Czech Republic. Contact: Erin Dolan, ASME, email: dolane@asme.org.


17. World Nuclear Exhibition, October 14-16, 2014, Le Bourget, Paris. Contact: Claire de Berny, Reed Expo, telephone: 33 1 47 56 24 09, email: Claire.deberny@reedexpo.fr.


Dry Storage Canister

EPRI has completed the initial set of inspections of loaded stainless steel dry storage canisters at three operational dry storage facilities. The goals of these inspections were to demonstrate the accessibility of the canister surface and to collect information on the environmental conditions experienced by critical canister elements over decades of storage, especially the welds. This work was partially funded by the U.S. Department of Energy through a contract administered by Idaho National Laboratory.

EPRI collaborated with Constellation Energy Nuclear Group (Calvert Cliffs), PSEG Nuclear (Hope Creek), PG&E (Diablo Canyon), Transnuclear, and Holtec International to obtain visuals, temperature measurements, and surface samples from the spent fuel storage canisters. While laboratory studies indicate that stress corrosion cracking can occur in the materials used for dry storage canisters when exposed to marine environments, data from in-service canisters are needed to evaluate such behavior in real-world conditions.

Due to differences in the geometry and design of the storage systems evaluated, the inspection techniques used for the first inspection (horizontal system) differed from those used for the second two (vertical systems). At Calvert Cliffs, a high-definition pan-tilt-zoom camera was lowered through the outlet of the horizontal storage module (HSM) for viewing the dry shielded canister surface and the support structure and interior surfaces of the HSM. Specially designed tools were inserted in the gap between the canister and the HSM for remote access to the canister surface. Tools included a thermocouple to measure the canister surface temperature, a dry scraping tool to collect surface samples directly from the canister surface, and a wet collection device that dissolved the surface deposits and collected them through a wicking device.

The systems at Hope Creek and Diablo Canyon took advantage of the vertical geometry by inserting a tool in the outlet vent near the top, lowering it into the annular gap between the canister and the concrete overpack to the desired location, and remotely actuating the tool to contact the canister surface. Similar dry scraping tools and a wet collection device were used for sample collection. Each tool had a thermocouple for temperature measurement and an onboard camera to visually examine the canister surface and ensure good contact of the surface and temperature devices.

The inspections revealed a coat of dust on the upper surfaces of all canisters, which was thicker for canisters in service the longest. One canister exhibited a few small surface rust spots on the base metal that were believed to be from free iron contamination, although no signs of rust were seen on any of the welds in the field of view. This is significant because the welds are more prone to stress corrosion cracking. The measured temperatures generally behaved as expected, dependent on initial heat load and location, with some differences between measured and predicted. Although there was a wide range in heat load (4–17 kW) and time in service (2–19 years), some areas of all canisters were in the temperature range for “deliquescence” to occur, which is when the salts on the surface absorb moisture from the air, potentially creating concentrated brines that can lead to corrosion. Measured temperatures are providing important benchmark information for best-estimate temperature calculations to more accurately predict when deliquescence may occur. Laboratory analysis of the collected surface samples indicate that the chloride concentrations are low and the cation/anion compositions are more representative of rainwater, indicating that the sites inspected are less marine, or corrosive, than anticipated.

Contact: Keith Waldrop, telephone: (704) 595-2887, email: kwaldrop@epri.com.

Source: Electric Power Research Institute’s (EPRI) Nuclear Executive Update, January 2014.

Refurbishment Specifications

EPRI has developed several guidance documents to assist nuclear plant owners in working with vendors to define and clarify refurbishment specifications for equipment such as pumps and motors.

Refurbishing equipment may be preferable to procuring a new replacement, particularly in cases where the equipment is obsolete and replacement with a different model would require significant resources. Long lead time and significant engineering effort is often needed to establish the suitability of a new item through activities such as prototype and equipment qualification testing. A carefully defined refurbishment specification can help assure a successful, on-schedule outcome.

The initial urge to attach the original equipment specification to a purchase order should be re-considered. Original specifications are not typically updated to reflect modifications and changes made by the plant subsequent to initial procurement and installation. In addition, original specifications do not always identify the quality oversight activities conducted by the purchaser during the original procurement, can include information not relative to a refurbishment, typically omit information that is relative to a refurbishment, and may include references to dated standards and obsolete materials.

Development of refurbishment specifications often requires active communication with the supplier to ensure the supplier understands the technical, quality oversight, and schedule requirements. Generic specifications developed by EPRI through the Large Electric Motor Users Group and the Pump Users Group for the refurbishment of medium-sized motors and pumps may provide a good starting point. Key considerations when preparing refurbishment specifications include:

- Provisions for design modifications and changes that were implemented after the component was provided in accordance with the original specification.
- Considerations for wear resulting from service (original dimensional information may no longer apply).
- Verification of clearances, specified tolerances, and alignment to ensure that the new parts mate and interface correctly with the original assembly or component.
Impact on seismic qualification, environmental qualification, and compliance with applicable construction codes.

Lessons learned through operating experience, as well as improvements in available material and manufacturing and inspection techniques.

Feedback mechanisms that can be used to monitor progress throughout the project such as status reports, document reviews, witness and hold points, tests and inspections.

Clarification of expectations for demonstrating compliance with out-of-date standards or specifications including reconciliation and customer approval requirements when applicable.

Requirements for obtaining customer approval prior to proceeding with repair techniques (such as peening to straighten a pump shaft) that may impact material properties.

Documentation requirements, including photographic evidence of inspections and work performed and as found and as-left reports.

Instructions for addressing non-conformances and deviations.

Participation in the oversight of refurbishment activities at the supplier’s facilities presents an excellent opportunity for new and experienced engineering and technical plant staff to become more familiar with the equipment for which they are responsible, particularly when they have not had the opportunity to become intimately familiar with the equipment through years of construction, start-up or operating experience.

Contact: Marc Tannenbaum, telephone: (704) 595-2609, email: mtannenbaum@epri.com.


**Alpha Monitoring Guidelines**

Minor defects in the fuel can release small amounts of alpha-emitting nuclides under certain conditions during a nuclear plant operating cycle. When these releases occur, they don’t represent a threat to public safety, but the alpha-emitting nuclides in the failed fuel elements can contaminate various areas of the plant. Because alpha-emitting nuclides are difficult to measure directly and can have significant dose impacts on workers, it is important to quantify their presence. EPRI’s alpha monitoring and control guidelines (3002000409) provide a risk-informed approach to monitoring based on the abundance of alpha particles as compared to beta and gamma species.

Alpha particles are typically produced in a nuclear power facility through neutron activation of uranium-238 (238U). Because greater than 95 percent of the uranium in power reactor fuel is 238U, substantial radionuclides are generated, including longer-lived alpha radionuclides such as plutonium-238, plutonium-239, plutonium-240, plutonium-241, plutonium-242, americium-241, neptunium-237, cadmium-242, and cadmium-244. The presence of these in the oxide films on material surfaces must be managed.

The controls applied to reduce beta-gamma contamination are often adequate to control alpha contamination. The EPRI guidelines, therefore, consider beta-gamma controls in determining when additional monitoring or controls may be needed for alpha contamination. This approach does not replace identification and monitoring for the alpha hazard, but allows for circumstances in which beta-gamma controls will be adequate to protect workers from both the beta-gamma and potential alpha contamination hazard.

EPRI has refined its monitoring and control guidance based on lessons learned from plant operating experience. For example, the revised guidance clarifies the characterization of radiological conditions, adds more specific work planning and work control guidance based on a risk assessment approach, and includes recommendations for additional bioassay monitoring when an intake is suspected. Recognizing that training is an important component of a radiation safety program, additional alpha-related training topics were included in the guidance.

Contact: Clay Perry, telephone: (202) 293-6184, email: clperry@epri.com.
1. In the United States, we are very fortunate to have the NRC, which has become a model for other regulatory bodies. There is some talk about an international safety organization, but there are also voluntary ways that people can learn from each other. Would you comment on the idea of global safety regulation?

Safety is first and foremost, and you’ll hear us talking about operational excellence around the world in everything that we do. AREVA is unique as a supplier in that we are members of the World Association of Nuclear Operators. As a member of that organization, we incorporate the unique perspective of nuclear operators at the corporate level.

We have four elements of operational excellence: safety, quality, performance and delivery. No matter where we are around the world, as a management team, we focus on all four of these pillars. When it comes to safety, we are talking about nuclear safety, radiological safety and industrial safety, and we have metrics that we regularly monitor at the board level for all of these items. We also focus on quality. We apply the most stringent standards, following not only the international standards, but also the local standards for quality. We also focus on performance and on-time delivery for our customers.

Importantly, we don’t just talk about these elements of operational excellence; we actually go out into the plants and into the field, and engage people in dialogue about them. When you talk about excellence as a culture, it’s about continually learning not only from the environment you’re in, but also from others around the world. It’s something you have to work on every day. It’s not something you can take for granted.

2. Cyber security has become a major concern in the utility industry and especially for the nuclear power plants. What is your approach to cyber security?

We offer unique cyber security practices and techniques to the industry through our partnership with cybersecurity industry partners. AREVA brings its digital expertise, plant engineering and system knowledge, and extensive regulatory experience to this partnership, while our industry-leading partners offer their expertise in providing cyber security support to U.S. defense, intelligence and civilian agencies.

Our approach is to look at the nuclear facility and also at the things that connect to the nuclear facility. We look at which areas represent the highest risks, and then our partners bring the monitoring and cyber technologies to bear in making sure that we have a robust defense system in place. We can also take that approach beyond the nuclear plant and expand it to protect the utility’s entire organization and assets.

3. In the United States, it seems to me that we are kind of slow on digitizing protective systems. AREVA was involved in digitizing systems at Duke’s Oconee station; what can be done to help utilities welcome a move to digital?

All three units at that station have upgraded their engineering safeguards and reactor protection system to AREVA’s digital safety-related Reactor Protection System and Engineered Safety Protection System. The first system has been installed now for nearly three years and day-to-day operations at the plant have become more efficient. It’s saving the plant on down-powers and transients due to human interactions. The system performs real-time surveillance now, so there’s no human error. I think, overall, the operation staff would tell you they really see a benefit to moving to the more modern systems. I would encourage other utilities to go see the system firsthand and talk to operators who are using the equipment day in and day out. That in itself—having that operating base here in the United States—will allow others to move forward.

Companies will have to consider digitization as they look at extending the lives of their plants from 60 to 80 years. The analog systems are already becoming increasingly difficult to find parts for and maintain, and the challenges of adding 20 years to a plant’s life will put old analog
systems into the forefront of the systems that need to be replaced.

4. How is AREVA keeping up with the application of latest technology to equipment and instrumentation?

You’re seeing different pieces of the instrumentation system get upgraded as new components get put forward. You’re seeing variable-speed drives being installed in different recirculation systems. You’re seeing smarter transmitters and relays being installed. This trend has been underway for some time when it comes to the smaller components, but now we’re talking about the larger systems and the integration of these systems.

Advanced technologies add value, in terms of safety. It makes the plant easier to operate. It gives the operator a better human interface and you can see the responsiveness of the plant much more clearly. As companies look at taking their plants from 60 to 80 years, you’ll see more and more plants looking to integrate these systems on a larger scale.

5. How do you think control systems will work 10 or 15 years from now? When the current generation is operating the plant—the ones who are always looking at their iPads and iPhones—what kind of control systems will they want?

We are already adapting our inspection equipment. For the younger generation of the workforce, game controller-like consoles run the robots, so the newer operators can maneuver the robot using a handheld device that they’ve grown up with. Our generation was using a stick with a button; they’re using both hands and all their fingers to maneuver the robot around and inspect things in minutes. These tasks took hours to complete before.

In general, as this next generation of skilled workers is coming in, I think the movement of technology into this industry and into the fleet ultimately will result in a safer environment, not only for the workers, but also for long-term operation.

We manage outages from a control center where we have live contact with all the outage crews that are doing anything, anywhere—from dry cask storage to fuel inspections. They’re all plugged in and communicating through this center. If there are any needs or items to follow up, they are immediately tied in with the engineers. In some locations, we also have smart boards, which allow us to conduct just-in-time training for the crews. So, it’s much more fluid now.

With current technology, we can also go into containment areas and, with the latest advancements in metrology, we can take a picture of everything. Before we go into a refueling outage, we can do all the movements and models to see how everything fits together. For modification, we can build simulations and actually move equipment around to find the safest and most efficient way of doing the job in terms of lifts and haul paths.

Aging concrete is also an issue that the industry is facing. We can take a picture of the concrete and map the cracks, and then we just keep taking pictures. Then you can tell if the crack grows or not. It’s that type of technology for managing aging that will enable us to keep these plants running more safely and healthily for longer periods of time.

6. You’re talking about predictive technologies. Is there a move in that direction by AREVA?

Yes. We have a number of tools including advancements in valve and concrete monitoring. We have made some investments in cable monitoring that we haven’t commercialized yet, but it’s getting there. In fact, in Lynchburg, Virginia, we’ve built the AREVA Solutions Complex. It’s a campus of eight different venues with laboratories where we can do all kinds of testing—seismic, environmental, radio-frequency interference, electromagnetic interference and metallurgical, just to name a few. We can do reverse engineering. We’ve got a tooling inspection area for fuel and we can test that in depth. We can look at either contaminated or noncontaminated equipment, in our refurbishment shops. So, we’re investing in a very large way into making sure that we can take customers not only from 40 to 60 years, but also advance and develop these technologies to go from 60 to 80 years.

7. Has any utility approached you for 80 years?

Yes, we’re in discussions with several. We’re doing our own internal studies as well about how we would approach going to 80 years.

8. We talked a bit about the younger generation. A lot of the older generation have already left the system. How you are maintaining the knowledge that these folks developed and passing it on to the younger generation?

There’s a lot of ways we do that. We have formal training programs and mentorships, but there’s nothing like doing the work, so we assign the younger workers to do tasks under the supervision of someone more senior who can pass on that knowledge. We try to do that in our work assignments and training programs throughout the country and for all of our different product lines.

Engineers these days are coming out much more sophisticated in terms of project management and communication skills. The type of learning that they’re doing is much more advanced. When we were in college, to get information, you would have to go to a library to get a book. If they didn’t have the book, it had to be ordered and you had to wait weeks for it. Well, now, it’s fractions of a second. They’re not spending the time in college looking for information; they have the information. The education that they’re getting is really about concepts. The younger engineers I see coming into the workforce are far more prepared and more well-rounded. What we focus on is getting them those direct experiences: making sure that they have access to lessons learned from the operating experience and databases, and that they’re applying those lessons in their early assignments.

9. Research is a lot easier now. Is it a total shift in culture?

It is. When we started, we didn’t have anyone to ask. We were figuring it out for the first time in the 1970s and 1980s. This generation not only has information instantly available, but they can also look at someone sitting beside them and say, “What did you mean when you wrote this?” It’s a paradigm shift and that gets back to the technology implementation that we were talking about. I think the technology is going to be an enabler in this industry, as it has been in so many
Sustainable, Safe Used Fuel Management – It’s not Waste until It’s Wasted
By Michael McMahon, AREVA TN

Michael McMahon
Dr. Michael McMahon is Senior Vice President of Columbia, Md. based AREVA TN Americas (a division of AREVA Inc.), the U.S. market leader in providing innovative total systems solutions for used nuclear fuel and radioactive waste storage and transportation. He was previously based in France and served as the U.S. director of international projects for AREVA’s Back End Business Group, which includes the group’s used fuel management, recycling, transportation and decommissioning activities.

Before joining DE&S, Dr. McMahon was an associate at the management consulting firm McKinsey & Company.

He is a graduate of the United States Naval Academy and has also earned a doctorate in nuclear engineering and an MBA from the Massachusetts Institute of Technology.

An interview by Newal Agnihotri, Editor of Nuclear Plant Journal at the NRC Regulatory Information Conference on March 12, 2014 in North Bethesda, Maryland.

1. What research and development is being conducted by AREVA or by the industry in fuel cladding degradation during dry fuel storage?

   The nuclear energy industry focuses on safety and incorporates “defense in depth” into the design of dry fuel storage systems. The integrity of the fuel cladding is important, but it is only the first line of defense. The second line of defense is the dry shielded canister (DSC). The DSC is a stainless steel shell with a basket inside that safely and securely holds fuel. It contains materials to make sure that the heat gets out, and it contains neutron poison materials to ensure it stays subcritical. Once the fuel is loaded into the DSC, the water is drained out and it is back-filled with dry helium. Then it is welded shut with two lids and transferred to the independent spent fuel storage installation (ISFSI). With AREVA’s technology, the DSC is then placed in a horizontal storage module (HSM), which is then bolted shut. This module serves as physical protection for the DSC and is a very safe, stable system.

   When it comes to the fuel, the material – in this case, the zirconium alloy cladding – is important. Our team evaluates how it behaves under radiation, heat and stress, and a lot of people are working with advanced cladding materials to see if they can reduce hydrogen pickup. As the fuel sits in the reactor, the cladding starts to absorb hydrogen, which can make the material more brittle. We need to make sure the material is strong enough so that it can be transported safely.

   On the operating fuel side, we’re working to keep the hydrogen uptake in the cladding low, which makes the material stronger and more durable. As long as the fuel cladding stays intact, it is a very safe system.

   The research we’re doing is innovating new technologies to improve safety, quality and cost and to ensure that the cladding is very robust and stays intact, that it can be transported safely, and that we understand the changes it will undergo over time in dry storage.

2. How is aging management of fuel in dry fuel storage accomplished?

   We are working directly with the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) on the high burnup fuel demonstration project. This project aims to take a dry cask with a bolted system, called a TN-32, and test it. We’ll put high burnup fuel into the canister and monitor the key parameters over a period of 10 or even 20 years. We want better long-term data for the behavior high burnup fuel in dry storage. For example, we want to study high burnup fuel with greater than 45 gigawatt-days per metric ton of uranium of burnup. As part of this project, we will take fuel samples from the canisters at specified times and send them to a laboratory to determine how the fuel responded to temperature and radiation.

   We’re also working on innovative projects related to gas monitoring. If the fuel in the canister develops a leak, it will emit a detectable gas. Our goal is to collect data to better understand what’s happening to the fuel as it ages in place in dry fuel storage.

   The fuel sitting in dry storage is in a fairly benign environment. All the water and moisture have been removed from the canister and it is filled with an inert atmosphere of dry helium to prevent corrosion. When it is transported, we have to account for vibratory stress and DOE is doing some analysis to see how the fuel responds. To address the behavior of fuel inside the canister, we have long-term monitoring and vibratory testing.

   We’re also evaluating the behavior of the canister itself for long-term aging management. We’re conducting research on chloride-induced stress corrosion cracking, which is cracking in metal that can occur under certain conditions, including the presence of chlorides and residual stresses in the material. For our canisters, we are looking at closure welds that seal the canisters, which can be a source of residual stresses, because different parts of the weld heat up at different rates. To relieve stress in a fabrication facility, canisters are placed in large ovens and heated slowly, which relieves the stress. However when
canisters are loaded at a customer site, the final welds are done in the field, so we need to find a different way to relieve stress. Since chloride-induced stress corrosion cracking also needs the presence of chlorides, we are looking at sites in coastal and marine environments, and conducting inspections and collecting data to better understand this phenomenon.

These actions are all undertaken to innovate new technologies and improve safety and performance while lowering costs.

3. What is AREVA doing to estimate the life expectancy of dry fuel storage?

We are actively developing inspection techniques that allow for the inspection and monitoring of these canisters for as long as they are in interim storage. The official license renewal is for 40 more years, but we’re working to address how long fuel will need to be stored, possibly 50 to 100 years. Further, the DOE has stated publicly that the earliest date for an operating geological repository for used nuclear fuel is 2048. If that date is correct, some of these systems will be 80 years old when the repository first opens.

We’re in the process of extending the license for the dry storage systems for an additional 40 years, similar to what is being done for the reactors, and we can extend it longer if necessary. It’s a very passive system. Regardless of the amount of time, it’s important to have effective techniques to monitor for signs of degradation and to make repairs, if necessary.

4. What is AREVA’s perspective on a long-term fuel disposal solution in the United States?

First, we need to focus on safety and sustainability. We need to demonstrate to the world that nuclear power is a viable source of sustainable energy. Our ability to manage the back-end of the fuel cycle effectively is a significant component to ensuring nuclear power’s sustainability. Thirty-five years ago, the United States chose to implement a once-through fuel cycle, but at AREVA, we advocate that all options be kept on the table. Recycling offers more options. The waste that comes out after recycling is vitrified and is designed to last millions of years in safe storage. We are strong advocates of recycling as an additional tool for the United States. This is not a technology issue; we already have the technology. It is a political and policy issue that will need to be addressed if we are actively going to address climate change. Nuclear energy will be needed for emission-free energy.

There are proven, reliable and economical technologies for recycling nuclear fuel today. These technologies provide the nuclear industry in France with many options for safely managing their used nuclear fuel. In the United States, we need to develop a long-term plan for managing the back-end of the fuel cycle.

Interim storage is an important tool, but it’s not a solution. It provides flexibility as we work toward a permanent solution.

In addition to developing a long-term solution for fuel disposition, we need to have a robust infrastructure in place for transporting used nuclear fuel. In France, the recycling facility in La Hague receives about 250 shipments of used nuclear fuel each year, which is equivalent to about one shipment each working day. This amounts to approximately 1,200 tons of used nuclear fuel.
Sustainable, Safe... (Continued from page 25)

fuel a year that AREVA transports. It’s a very reliable and robust system with very safe containers. The United States needs to start developing that robust transportation infrastructure now because regardless of the eventual solution for used fuel management, we will eventually need to safely transport used fuel. Transportation is one of the most secure parts of the fuel cycle and we need to effectively communicate about this process to the public in order to increase public acceptance.

5. What is holding U.S. policymakers back from the recycling option?

When the United States decided not to pursue recycling in the late 1970s, the country was concerned about nonproliferation. Policymakers believed that if we didn’t recycle, nobody else would. Further, they did not believe that this decision would have a significant negative impact on the U.S. nuclear industry.

Now, nearly 40 years later, it’s time to revisit this decision and the logic behind it. No other countries followed our example and stopped recycling their nuclear fuel. While Germany did shut down its domestic recycling plant, it sent its used nuclear fuel to France to be recycled.

In terms of non-proliferation, there are now more declared nuclear weapons states in the world today than when the U.S. announced its ban on recycling. Notably, none of these countries obtained weapons material by recycling commercial light water reactor fuel. Finally, the lack of a permanent solution for used fuel has had a huge negative impact on the U.S. nuclear industry with more than $10 billion of ratepayer money wasted and waste confidence rulemaking vacated by the U.S. Court of Appeals. The policy hasn’t changed because policymakers appear to assume that the world hasn’t changed in 40 years, but it has. If people had the same attitude about technology, everyone would still be using computers that cost millions of dollars and fill up an entire room instead of carrying them in their hands.

We must challenge the conventional wisdom about recycling and prevent misinformation that has prevented us from taking a serious look at that option.

6. Which countries are recycling, in addition to France?

The UK, France, Japan, Russia and India all recycle. China has also chosen to recycle, but its program is still relatively young. They haven’t built a commercial facility yet, but they are committed to a closed fuel cycle. Every country with a large ongoing nuclear program, except the United States, has opted for a closed fuel cycle.

The few countries that have implemented a direct disposal solution, such as Finland and Sweden, have relatively small nuclear programs. For a small nuclear program, an open fuel cycle can work. Finland has only four reactors and they’ve got a lot of territory relative to the size of their population. Sweden has about 10 reactors. However, for countries with large fleets operating over a long period, the issue of directly disposing of used nuclear fuel can be more challenging.

We still need a geological repository with recycling, but the volume of waste in the repository would be much smaller. Further, the waste is vitrified and there is research showing that glass is incredibly durable. It will remain intact for longer than it will take for the nuclear waste to decay to the level of the uranium that was originally taken out of the ground. This is a very safe and sustainable system, but unfortunately, recycling of safe used nuclear fuel just hasn’t gotten traction in this country.

7. Do you have any concluding comments?

Safe, sustainable and responsible management of the fuel cycle demonstrates that nuclear power is a sustainable energy. We need to innovate new technology and find a long-term solution for used nuclear fuel, which will help pave the way to build more new plants and increase public acceptance that nuclear energy is a sustainable energy source for the future. As a result, the effective management of the back-end of the fuel cycle should be a priority for the future of safe, reliable, affordable, low-carbon nuclear energy.

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Leading the... (Continued from page 23)

other industries, enabling us to become more efficient at what we do and at the same time, helping us to provide higher-quality solutions.

10. I’m looking forward to the day construction starts on an EPR™ in the United States. How is the lead plant doing in terms of US NRC design certification?

We’re building four EPRs around the world right now and hopefully Hinkley Point in the U.K. will go forward, so that will be six. Here in the U.S., our design certification is well underway. We have a work schedule laid out with the NRC and we’re into the final portion around how to close chapters. And with that, we’re working through the final issues with comments on how our instrumentation and control system is applied in the EPR, but we feel very confident with the design.

11. Which utility is the lead on the combined operating license application?

It’s UniStar and PPL. UniStar is the lead and PPL is right there with them. It’s PPL on Bell Bend in Pennsylvania and UniStar on Calvert Cliffs Unit 3 in Maryland.

12. Anything you’d like to add?

In the vein of technology that we talked about earlier, I think AREVA is embarking on quite an innovative approach to the industry worldwide. You’ll see the advent or the implementation of the advanced technologies and systems we talked about today, not only to take the near-term plants to the next level, but also to extend operation not just to 60 years, but to 80 years. And you’ll see these technologies in the newer plants evolve to even safer designs. We’re very excited about the types of forward-looking work AREVA is doing in this area.

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HTGR Reactor
By Lewis Lommers and Farshid Shahrokhi, AREVA Inc.; Chris Hamilton and Matt Richards, Ultra Safe Nuclear; Scott Penfold, Technology Insights; John Mahoney, High Expectations International, LLC.

Lewis Lommers
Lewis Lommers leads High Temperature Gas-Cooled Reactor engineering at AREVA Inc.

He has over 25 years of experience working on HTGRs in various areas including system design, transient analysis, performance optimization, and overall concept development. His experience covers several High Temperature Gas-Cooled Reactor (HTGR) concepts including the Modular High Temperature Gas-cooled Reactor (MHTGR), Gas Turbine Modular Helium Reactor (GT-MHR), ANTARES (an indirect cycle HTGR concept that AREVA developed internally from about 2002 to 2007), and Next Generation Nuclear Plant (NGNP).

He has a Bachelors degree in Mechanical Engineering from the University of Washington and a Masters degree in Nuclear Engineering from Purdue University.

Responses to questions by Newal Agnihotri, Editor of Nuclear Plant Journal. The responses are for the HTGR concepts being promoted by the NGNP Industry Alliance Limited and its members. They are not necessarily reflective of the opinions or positions of the US Government’s NGNP Program.

1. What is the cost in mills (1/10 of the US cent) per kilowatt hour for producing electricity with a NGNP’s HTGR reactor? Provide the following breakdown: maintenance cost, operation cost, and fuel costs.
   - Fixed O&M (w/o Fuel) -14 mill/kWh
   - Variable (Inc. Fuel) - 11 mill/kWh
   - Total - 25 mill/kWh
   *Note: This does not include capital or other project costs.
   Costs are for an electric only plant. For cogeneration applications, costs will vary depending on specifics of the application.
   This is not an all-in production cost or a means to compare technologies on cost since capital investment and investment return has not been described in the responses.
   The Alliance has done extensive economic analysis to characterize the market and associated economics. We have developed an commercialization strategy and an enterprise architecture that identifies capability and costs associated with: a development venture; a deployment project; infrastructure framework; technology expansion that includes lesson learned, operational experience (OE) and advanced materials research and development; and activities for program direction anticipated to support maturing technology application, on-going economic analysis and providing insight into additional markets and opportunities. Estimated costs to complete each of the Enterprise activities are listed in the updated business plan.

2. What improvements have been made to the instruments within the Reactor Pressure Vessel (RPV) as compared to current reactor technology for measuring pressure, temperature and other parameters?
   Significant improvements in instrumentation technology are not required for the Steam Cycle High Temperature Gas-cooled Reactor (SC-HTGR) design. We have eliminated the requirements for extensive incore instrumentation. The primary thermal measurements are limited to inlet and outlet coolant temperature and the coolant flow rate. Neutron detectors for the reactor protection system are located outside the reactor pressure vessel (RPV). A traversing flux monitor is used periodically to measure the axial power profile.

3. What diagnostic mechanisms have been built in to monitor the degradation of material, cables, equipment and instruments within the RPV?
   The SC-HTGR has ceramic core. There are no cables in the core. The metallic components in the reactor pressure vessel are located in areas that are only exposed to the cool core inlet temperature and are designed to retain their mechanical properties for the life of the component.

4. What prototype testing has been done to validate the design to protect problems during construction and operation of the NGNP’s HTGR reactors?
   All key equipment, including the reactor core, the helium circulators, the steam generators, and the vessels are based on technology which has been proven either in previous HTGR projects or other industrial applications. During final design and pre-fabrication development, mockups will be used as appropriate to confirm configuration and fabrication approach. Confirmation testing of components or component modules will be performed as necessary prior to installation.

5. Who will be responsible for refueling the reactor? Will it be the nuclear power utility or the manufacturer?
   Refueling will be performed by the utility personnel. SC-HTGR refueling is accomplished with remote mechanical refueling machines based on the concept proven successful at Fort St. Vrain. Specific fuel movements are controlled by the automated system according to a preplanned sequence developed prior to each refueling campaign. All movements are monitored from the refueling control station, and all movements are recorded and reviewed to confirm the accuracy of the refueling operation.
6. What are the plans to repair possible problems within the pressurizer, and other equipment and instruments within the RPV?

SC-HTGR does not have a pressurizer. All major components (core barrel, core support structure, permanent graphite reflector, steam generator, etc.) are designed for the plant lifetime. Nonetheless, all components within the primary coolant vessels are replaceable, if necessary. Steam generator performance in gas-cooled reactors has historically been very good. If necessary, individual tubes can be plugged easily. The entire steam generator can also be replaced, although this is not expected to be necessary. Helium circulators will be removed and replaced periodically for routine refurbishment.

7. How long can the reactor keep operating without:
   • Off-site power supply?
   • Cooling water supply?
     **Off-site power supply:**
     Offsite power supply is not an issue for normal reactor operation.

     For accident response, off-site power is never required. No electrical power is required for heat removal or for activation of any safety systems. Limited onsite electrical power is needed only for accident monitoring, in order to inform regulators and to provide data necessary to approve plant restart.

     **Cooling water supply:**
     During normal reactor operation, cooling water is used for normal heat rejection from the condenser as in any other thermal power plant.

     For accident response, the reactor can go seven days before replenishment of cooling water in the Reactor Cavity Cooling System (RCCS) is needed. Going beyond seven days without adding water could result in equipment damage, making plant restart questionable. But public safety would still be maintained.

8. Are there provisions to bring external portable connections for insuring water supply and electric connection for cooling in case of a Beyond Design Basis Event (BDBE)?

   No electrical connections are needed for safety (connections will be provided for monitoring). No water makeup is required for safety for Beyond Design Basis Event (BDBE). Nonetheless, provisions are available for replenishing the RCCS.

9. What enhancements have been made to the NGNP’s HTGR design after Fukushima?

   No enhancements were necessary. SC-HTGRs can withstand extended power outages. The passive and intrinsic safety of the plant will maintained by the initial design.

   The plant will be designed to withstand the local maximum expected seismic activity and any other external effects (tsunami, tornado, etc.).

10. Provide the capacity for different applications including hydrogen generation, electricity generation and district heating for a NGNP’s HTGR reactor?

   The reference 625MWt modular SC-HTGR selected by the Alliance for initial commercialization operates at a conservative reactor outlet temperature of 750°C producing steam at 16.7MPa/566°C comparable to modern fossil boilers and gas-turbine combined cycle power plants. Combined with safety characteristics that allow close-in siting, the HTGR can serve a range of applications similar to its fossil counterparts, including electricity generation, process steam and medium-range thermal energy for industrial applications (e.g., petrochemical, heavy oil and bitumen recovery from oil sands via steam injection, as well as lower temperature applications such as desalination and district heating). With its higher-temperature capability and efficiency, the HTGR is particularly adaptable to site conditions requiring dry cooling.

   In the future, very high temperature versions of the HTGR with increased outlet temperatures around 950°C will be applied to more advanced applications (e.g., direct hydrogen production via thermochemical water splitting and direct heat supply to high temperature chemical processes such as ethylene cracking). The HTGR coated particle fuel system has already been demonstrated to be viable for such operating temperatures in test reactors. However, further advances will be necessary related to available materials for heat exchangers to transfer the thermal energy from the HTGR to the application process. Initial commercialization of the SC-HTGR will greatly facilitate the evolution to these future advanced systems.

11. What is the design life of the plant 60 years or 80 years?

   The present design basis for the modular HTGR is 60 years, which is consistent with both investment guidelines and the available design basis for materials that operate above the time-independent temperature regime. With appropriate reevaluation and/or replacement of a limited number of high-temperature internal components, extended lifetimes beyond 60 years should be possible. Neutron irradiation effects on the primary pressure boundary components are typically lower in HTGRs. The plant’s non replaceable components (RPV) are made from current pressurized water reactor (PWR) reactor vessel materials. So subsequent life extension work underway for PWRs will be used to support 80 year life. All internal components are replaceable.
HTGR Reactor...
(Continued from page 29)

12. How often the plant refueling will be needed?

½ of the core every 18 months.

13 You may share any domestic or international utilities who have expressed interest in NGNP’s HTGR?

The design will deployed for industrial process heat usage. The plant will be operated by a nuclear process heat utility operator. Several utility operators are interested in many have existing customers in industrial settings that are interested in least cost energy options. The Alliance is focused on industrial end-users and not just utilities. Entergy Corporation, a utility headquartered in New Orleans, Louisiana has been an active member of the Alliance and has participated in studies supporting viability of deployment projects for industry. Over the past five years, the Alliance has met with most U.S. Nuclear utilities who are tracking the development of the technology and have interested industrial users as customers.

Internationally, the Alliance is collaborating with Korean Nuclear Hydrogen Alliance (see http://energy. korea.com/archives/49866). It has endorsed work on the High Temperature Test Reactor (a prismatic block HTGR) in Japan for the Japan Atomic Energy Agency (JAEA), and is working with the Nuclear Cogeneration Industrial Initiative (NC2I) European Alliance (see http://www.nc2i.eu/) to form an agreement that is mutually beneficial to our mission of commercializing HTGR technology.

14. How have the lessons learnt from Fort St. Vrain nuclear power plant been incorporated to ensure the technological success of NGNP’s Small Modular Reactor?

While Fort St. Vrain (FSV) experienced a number of issues consistent with its status as a demonstration plant, the most impactful issue was water ingress through the novel, aggressively designed water-lubricated bearings for the circulator which required maintaining a delicate pressure balance between the lubrication system and the primary circuit helium.

While none of the issues posed a safety concern, the water ingress was a major operational problem, resulting in long outages to remove the water. In contrast the preceding Peach Bottom I HTGR used more conventional, electrically driven circulators with oil-lubricated bearings and did not experience water ingress problems.

The SC-HTGR uses electrically driven circulators supported on magnetic bearings eliminating the main source of water ingress at FSV. In addition, provisions are made to drain liquid water from the primary vessels in the unlikely event if it enters from another source.

A direct result of the water in FSV was corrosion and adverse interaction with internal lubricants such as in the control rod drives. Such locations offer a cold trap where water can collect. In the SC-HTGR design, such sensitive locations are purged with clean dry helium to prevent the buildup of moisture or other potential contaminants.

Another lesson related to FSV concerns with the fuel cycle. FSV used a thorium/HEU fuel cycle. While neutronically optimum, the highly enriched uranium (HEU) cycle is costly due to security concerns associated with the manufacturing and transportation of fuel. The SC-HTGR uses a low enriched uranium (LEU) cycle, eliminating the HEU issues.

All of the FSV “lessons-learned” are being addressed in the SC-HTGR detailed design. But the points discussed above were the most important.

15. Provide a brief description of the NGNP’s HTGR by providing additional information which is not covered above.

Today, process heat requirements for energy-intensive industries around the globe are provided almost entirely by fossil fuels. Electrical power for these industries is provided primarily by combusting solid, liquid and gaseous fossil fuels. Consequently, these industries are hostage to environmental concerns, unpredictable government policies, uncertainty of supply and price volatility. Modular HTGR nuclear technology provides an important option that addresses these issues. It provides process heat at the temperatures needed by industry and power with competitive economics, compelling safety, and minimal environmental concerns.

For markets reliant on premium fossil fuels, commercializing the HTGR makes available the only game-changing technology on the horizon that can address global energy policy goals of energy and feedstock security, economic growth (jobs) and carbon footprint. Trends in fossil fuel prices suggest that modular HTGR technology integrated with carbon conversion technologies provides an economic approach to production of synthetic transportation fuels, chemical feedstocks and chemicals with a minimal carbon footprint.

During both normal operation and under accident conditions no explosive gases are produced by the fuel materials or core infrastructure — the materials were selected and designed to preclude this. Used nuclear fuel from a HTGR requires no cooling water or active systems for storage or heat transfer, relying instead on natural convective air flow.

The safety case has been demonstrated in the German Arbeitsgemeinschaft Versuchsreaktor (AVR) HTGR and recently in the 10 and 30 MW designs in China and Japan respectively. In those tests, self-limiting reactor shutdown was demonstrated by stopping normal operation forced cooling of the helium. The reactor heated up to a safe temperature where the nuclear fission reaction intrinsically shut itself down without any damage to the reactor and without any release from the ceramic coated fuel.

Nearby public and industries need not shelter or evacuate for any event challenging reactor safety. This allows a close-in siting capability needed for process steam/heat loads, plus anticipated improved public and investor acceptance. Long-term investment risk due to safety concerns is minimized for both the reactor plant itself and for collocated industrial facilities.

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Nearly all of the approximately 100 commercial nuclear power plants in operation in the United States today began operation in the 1970s and 1980s. Along with the new nuclear plants that will be coming on line in the next few years, the existing commercial nuclear fleet represents a substantial national resource, providing approximately 20% of the nation’s electrical power, and doing so without contributing to climate change and without emitting any air pollution. These plants also provide an extraordinarily reliable source of baseload electrical generation, with each nuclear plant producing a steady output of power, on average, more than 90% of the time. There are regulatory proposals underway at the U.S. Nuclear Regulatory Commission (NRC) that could present business planning challenges for licensees considering the long-term operation of the domestic commercial nuclear fleet by undercutting the stable and predictable regulatory framework for license renewal.

The NRC initially licensed each existing nuclear power plant to operate for 40 years under the Atomic Energy Act of 1954 (Prior to 1975, nuclear power plants were licensed by NRC’s predecessor, the Atomic Energy Commission). This time frame was selected for economic and antitrust reasons rather than for any technical or safety limitations on the continued operation of a nuclear facility. Operating licenses can be renewed for up to 20 additional years, and there is no statutory or regulatory limit on how many times an operating license can be renewed. As of today, nearly three-quarters of the 100 operating reactors have obtained renewed NRC operating licenses. Most, if not all, of the remaining plants have applied or are expected to apply for renewed licenses.

But even with a first license renewal, the licenses for the oldest plants will begin to expire in 2029, with most of the remaining operating licenses expiring in the 2030s and 2040s. Absent the contribution of many new reactors—or without a second or “subsequent” license renewal for existing reactors, the United States will lose this safe, clean, carbon-free, and reliable baseload electrical power resource. This may seem far in the future, but the regulatory framework for subsequent license renewal is being evaluated now.

Under current conditions, lost nuclear generation would largely be replaced by fossil-fueled power sources, such as natural gas. This is primarily because renewable energy sources, such as solar and wind, are not baseload electrical generating sources. The loss of nuclear baseload generation and replacement with fossil-fueled sources would, among other things, eliminate the greenhouse gas emissions benefits provided by the existing nuclear fleet. In addition, as our reliance on fossil fuel grows, we would lose the reliability and diversity-of-supply benefits provided by the existing nuclear fleet. Given the long lead times associated with baseload energy planning and approvals, we need to begin laying the foundation now to retain existing nuclear assets beyond the next few decades.

A Well-Established Regulatory Process Is Already in Place

The NRC has a mature and tested regulatory process for the renewal of nuclear power plant operating licenses. This regulatory process is outlined in Title 10 of the Code of Federal Regulations (CFR), Part 54. These rules have been in place since 1995 and have been the framework for the successful renewal of 73 licenses to date. Numerous plants have already begun operating beyond their original 40-year licenses, and have successfully implemented the additional programs required to ensure plant safety during the period of extended operation. The Part 54 rules explicitly allow for and were designed to consider subsequent license renewals. Thus Part 54 can and should govern requirements for applications and the NRC’s granting of subsequent license renewals. This is because the methods licensees use to
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manage age-related degradation during the first license renewal term can continue to ensure safety during a subsequent license renewal term.

The NRC’s rules and safety requirements for license renewal in Part 54 are based on two fundamental principles. The first principle is that the existing regulatory process governing all operating plants in 10 CFR Part 50 is adequate to ensure that the plant-specific licensing basis for all plants provides and maintains an acceptable level of safety. The second principle is that the plant-specific licensing basis established under Part 50 must be maintained during the license renewal term in the same manner and to the same extent as during the original licensing term (See “Final Rule, Nuclear Power Plant License Renewal; Revisions,” 60 Fed. Reg. 22,461, 22,464 (May 8, 1995). The only additional safety measures necessary for license renewal are those required to manage the effects of age-related degradation in long-lived structures, systems, and components that perform or support the performance of certain functions defined in the Part 54 regulations during the period of extended operation. These principles apply equally to both first and subsequent license renewals.

The Part 54 rules focus on aging effects, as manifested in degraded performance or material condition over time, rather than the underlying aging mechanisms. Aging effects, such as cracking, embrittlement, or wall-thinning of components, can be monitored to identify the need to take corrective actions at the appropriate time.

The focus on aging effects ensures that the NRC’s safety review primarily evaluates whether applicant programs are adequate to monitor performance and correct age-related degradation in a timely manner. Furthermore, consideration of measureable aging effects, rather than theoretical or particular observed mechanisms, ensures that the regulatory process does not transform into an “open-ended research project,” which the NRC sought to avoid when it created the license renewal rule. Research into materials, aging mechanisms, and aging management techniques will not stop. The industry can and will continue to improve its aging management techniques based on both laboratory research and operating experience. However, emphasizing measurable aging effects allows licensees to develop effective aging management programs based on existing research and operating experience. The Part 54 rules already provide a framework that allows licensees to take advantage of improvements over time, based on new research and operating experience.

Aside from the robust technical nuclear safety review process established in Part 54, the NRC has a well-established environmental review process for license renewal. Recognizing that the environmental impacts of license renewal are much smaller than for initial construction because the more significant impacts of constructing the plants have already occurred, and that the environmental impacts of continuing to operate for 20 additional years are often similar across plants, the NRC Staff has developed a Generic Environmental Impact Statement (GEIS). This GEIS evaluates the environmental impacts of license renewal on a generic basis for the vast majority of issues. The GEIS and its associated rules relegate the remaining impacts for evaluation on a site-specific basis. As part of each license renewal review, the NRC evaluates those site-specific environmental impacts, considers new and significant information related to the generically evaluated impacts, and publishes a supplement to the GEIS for that plant. The GEIS is periodically (Continued on page 34)
updated at approximately 10-year intervals, and was last revised in 2013, so it is up to date. This well-established process can also be used to evaluate the environmental impacts of continuing plant operations for a second 20-year period.

The NRC Staff’s Proposals

In January 2014, the NRC Staff prepared an options paper for consideration by the five NRC Commissioners (SECY-14-0016 – Ongoing Staff Activities to Assess Regulatory Considerations for Power Reactor Subsequent License Renewal (Jan. 31, 2014), available at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2014/2014-0016scy.pdf). Overall, the options paper reaffirms the existing framework in Part 54 as the regulatory basis for subsequent license renewal, concluding that “the license renewal process and regulations are sound and can support subsequent license renewal.” “Option 1” would involve no rule changes. However, the paper recommends clarifications and updates to the license renewal rules—labeled Options 2 and 3. These changes would apply to both first and subsequent license renewal applications and would, for example, update Part 54 to cover certain post-1995 rule changes in other areas or codify already-established NRC Staff guidance.

The options paper recommends what is labeled “Option 4.” This recommendation includes the Options 2 and 3 changes, and would add more substantive rule changes. First, the Staff proposes additional requirements for applicants to report aging-related degradation to the NRC and to demonstrate the effectiveness of aging management activities. Second, the Staff would reduce the time window within which an applicant may submit a subsequent license renewal application before its existing license expires, from the existing 20 years to a shorter period, so, in the Staff’s opinion, applicants have adequate aging management experience before they submit subsequent license renewal applications. And third, the Staff expresses the concern that certain site-specific characteristics, such as severe weather and external hazards, may change over time as a result of climate change. Therefore, the Staff proposes to verify the validity of certain original design input parameters, primarily through an update to Part 50 requirements (i.e., outside of the license renewal process). But, if necessary, the Staff’s paper suggests that it may seek to consider the adequacy of these aspects of a plant’s licensing basis as part of the subsequent license renewal review. If the Staff proceeds with rulemaking, it currently envisions publishing a proposed rule in 2016 and a final rule in 2017, as the first application for subsequent license renewal could be submitted as early as the end of 2017 or beginning of 2018.

This rulemaking proposal is unnecessary and has the potential to undercut the stable and successful regulatory framework in Part 54. The clarifications and updates that the Staff proposes in Options 2 and 3 can easily be addressed through Staff guidance, outside the rulemaking process. Indeed, some of these proposed changes are already covered in guidance, and others address unusual situations that will likely affect very few plants, so there is no need for a new across-the-board rulemaking. As to Option 4, aging management effectiveness can and should be verified through existing processes, as envisioned in the original Part 54 rules. Likewise, there are existing requirements to report age-related degradation. And the industry has initiatives under way to enhance the guidance on self-assessments of aging management program effectiveness and on the reporting of operating experience.

The proposal to reduce the time window for applications, which is part of Option 4, is unlikely to lead to any enhancement in safety. Most aging management programs are already in place at plants before the first license renewal, and applicants for subsequent license renewal will have accumulated more operating experience with existing programs than first license renewal applicants. Moreover, as previously noted, many plants throughout the country are already operating beyond 40 years and have implemented their remaining aging
management programs. Most aging management programs are common across the industry; therefore, by 2017 or 2018, when the first subsequent license renewal applications are expected to be filed, there will be considerable operating experience available for subsequent license renewal applicants to draw upon. Thus, from a safety perspective, the existing time window for applications is more than adequate to ensure that sufficient operating experience is available for subsequent license renewal applications.

Moreover, the Staff’s proposal presents potential business planning challenges for utilities, particularly under current market conditions. While SECY-14-0016 suggests that business planning “can still be accomplished,” there is no safety basis for imposing new business planning restrictions on plant owners who must plan for the future of the commercial nuclear fleet. Any rulemaking will introduce uncertainty and delay for the first set of subsequent license renewal applicants. First-of-a-kind subsequent license renewal applications typically require considerable time to prepare, so the regulatory requirements should be in place well in advance. The NRC’s establishment of a more restrictive business planning window could lead owners of safe and otherwise economically-viable nuclear plants to replace those assets with fossil-fueled generation.

Finally, the continuing validity of site characteristics and design parameters at a plant is a question that should be handled as part of the NRC’s ongoing oversight of plant operations, which is outside of the license renewal process. Under the first principle of license renewal, if these parameters must be reevaluated to ensure that the licensing basis for all plants provides and maintains an adequate level of safety, then this must be done as part of ongoing regulatory oversight under 10 CFR Part 50, not during the license renewal review.

The Road Ahead

The NRC Commissioners are now considering the NRC Staff’s proposals. The decisions made in the coming months will determine whether the NRC carries forward the well-established, robust, stable, and successful regulatory framework for license renewal, or embarks upon a new rulemaking, with the attendant introduction of uncertainty and delay while the regulatory process is in flux. Under current energy market conditions, the NRC’s decision could make the difference for the future of our nation’s cleanest and most reliable source of electrical power.

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Vogtle Unit 4

Georgia Power recently announced completion of the latest major milestone in the construction of Plant Vogtle units 3 and 4 near Waynesboro, Georgia. On May 8, 2014 the project team successfully placed the Unit 4 containment vessel bottom head (CVBH) into that unit’s nuclear island. The CVBH weighs more than 1.8 million pounds, or 900 tons, and is nearly 38 feet tall and 130 feet wide. The component consists of dozens of individual steel plates and was fabricated on site by CB&I, the project’s contractor. The CVBH was lifted into place using a 560-foot tall heavy lift derrick, one of the largest cranes in the world, and took approximately four hours. The placement of this component is the latest illustration of the continued progress of the construction of the two new nuclear units which use state-of-the-art AP1000 technology and are among the first new nuclear units to be built in the United States in 30 years. Once the new units come online, Plant Vogtle will be the only four-unit nuclear facility in the United States.

Since the beginning of 2014, the project has marked several other major milestones including placement of the 460-ton CR10 module (or cradle) into the Unit 4 nuclear island in February 2014 and the 2.2 million-pound CA20 module for Unit 3 in March. The CR10 module, which resembles a concave bowl with a hollow center, is the structure upon which the CVBH rests.

Increased efficiency is being achieved throughout Unit 4 construction and can be attributed to the successful implementation of lessons learned from previous construction on Unit 3. Additionally, the initiation of the Operations Control Center (OCC), an on-site facility staffed 24 hours a day, is providing a central point for reporting, analysis and resolution of project challenges. The meeting of major Unit 4 milestones such as placement of basemat rebar, basemat concrete, and the CR10 module are direct results of both the OCC and implementation of lessons learned.

Vogtle Unit 3

On Saturday, March 8, 2014, the project team successfully placed the CA20 module into the Unit 3 nuclear island. Weighing more than 2.2 million pounds, or 1,100 tons, and towering more than five stories tall, the module is the heaviest “lift” of the project to date. With a footprint of approximately 67 feet long by 47 feet wide, the critical module will house various plant components, including the used fuel storage area.

2.2 million-pound CA20 module placed into Vogtle Unit 3 nuclear island

Including assembly activities at both CB&I’s Lake Charles facility and onsite at Plant Vogtle, the module was assembled from prefabricated wall and floor sections and transported to the site by rail and truck for placement. It was lifted into place using a 560-foot tall heavy lift derrick, one of the largest cranes in the world.

Visible progress continues to be evident for both units 3 and 4, which use Westinghouse’s state-of-the-art AP1000 technology and are among the first new nuclear units to be built in the United States in 30 years. Since Jan. 1, 2014 the project has marked several other major milestones including the February, 2014 placement of the 460-ton CR10 module (or cradle) into the Unit 4 nuclear island.

The Vogtle 3 and 4 expansion is part of Georgia Power’s long-term, strategic plan for providing safe, clean, reliable and affordable energy for Georgians over the next 60 years. On Feb. 28, 2014 the Company filed the combined 9th and 10th Vogtle Construction Monitoring (VCM) Report with the Georgia Public Service Commission (PSC) which reports that the construction of the new units is progressing well with both units scheduled to begin commercial operation by the end of 2018. The report also recapped 2013 project milestones including the placement of nuclear concrete for both units 3 and 4 and the placement of the CVBH and reactor vessel cavity (CA04) for Unit 3. Once the units enter service, fuel efficiencies from nuclear generation combined with ongoing customer benefits such as the recently finalized federal loan guarantees, etc.

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are expected to put downward pressure on customer rates – cementing the project’s status as the most economic choice for meeting Georgia’s future energy needs.

The project is the largest job-producing project in the state, employing approximately 5,000 people during peak construction and creating 800 permanent jobs when the plant begins operating.

Southern Nuclear, a subsidiary of Southern Company, is overseeing construction and will operate the two new 1,100-megawatt AP1000 units for Georgia Power and co-owners Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia and Dalton Utilities. Georgia Power owns 45.7 percent of the new units.

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The 1,100-ton CA20 module is placed into Plant Vogtle Unit 3 nuclear island. The module, which is more than five stories tall, will house various plant components, including the used fuel storage area. March 2014.
Overview

The NSSS (Nuclear Steam Supply System) integrity monitoring system is the only one to provide the structural health monitoring for the reactor pressure boundary components on an on-line basis. Conventionally, it is comprised of four independent sub-systems such as IVMS (Internal Vibration Monitoring System), LPMS (Loose Part Monitoring System), ALMS (Acoustic Leakage Monitoring System), and RCPVMS (Reactor Coolant Pump Vibration Monitoring System).

The IVMS is designed for early detection of the degradation of the preload condition of the reactor internal structures. The reactor internal structures consist of core support barrel assembly, upper guide structure, core shroud assembly, low support structure, and hold down ring. They are subjected to flow-induced vibration due to the high pressure reactor coolant flow. The flow-induced vibration of the core barrel assembly may cause the degradation or loss of the axial preload at the upper support flange in the reactor pressure vessel. It can also result in the loosened or detached parts inside the reactor vessel. This might cause significant core damage or coolant flow blockage in the fuel channel. Thus IVMS is mainly monitoring the change of vibratory modal frequencies of the core barrel assembly to early detect the degradation of the axial preload by using ex-core neutron noise signal.

The presence of a loosened or detached metallic loose part within the reactor pressure boundary can give rise to the degradation of the reactor system’s structural integrity. That is, not only it may result in the mechanical damage and fretting wear due to its repeated impact on the system, but also can cause a partial flow blockage inside the fuel channel, a potential for control rod jamming, and the accumulation of radioactive substances in the primary system. Thus the primary purpose of the LPMS is to detect the presence of a metallic object within the reactor coolant system using the vibration sensors installed on the outer surface of the system. Ultimately, it should give as useful information as possible to identify the loose parts such as discrimination from false detection, localization of the object and estimation of the structural effect on the pressure boundary due to its impact.

The primary purpose of the ALMS is to monitor coolant leakage in the potential leak regions such as the reactor vessel, welded region in pipings, and valves, etc. The second purpose is to detect initiation of crack on the surface of the pressure boundary of the reactor coolant system. The leak detection is very important since the leakage could cause a loss of coolant accident. The ALMS normally consists of two subsystems. The first one is for monitoring the opening status of the PSV (Pressurizer Safety Valves) by the flow through the valves which had been mandatorily recommended by U.S. Nuclear Regulatory Commission Guide 1.45 (Reactor Coolant Pressure Boundary Leakage Detection Systems). It is called as PSV system. The second one is for detecting leaks and cracks in the reactor coolant system pressure boundary at specified sensor locations. It is called as non-PSV system, where AE (Acoustic Emission) technique has been used to detect the stress waves caused by the occurrence of a crack in a solid structure and the occurrence of fluid leakage.

The primary function of the RCPVMS is to monitor the shaft displacement and the rotational speed of a reactor coolant pump shaft and to monitor the vibration level of the RCP frame. The RCPVMS is designed to provide an alarm signal to the Main Control Room when the vibration level exceeds the allowable limit. It also provides diagnostic information to be used in analyzing the status of the RCP frames and their shafts, detecting the abnormal symptom of the shaft crack, and adjusting the RCP alignment and rotor balancing. In this system, two types of sensors are used. One is an accelerometer and the other is a proximity probe. Typically three accelerometers are mounted on the RCP frame to measure the horizontal and axial vibration levels of the RCP frames. Three proximity probes are mounted around the RCP rigid coupling. Two of them are used to measure orbit (path of shaft centerline motion) and the vibration level (displacement) of the RCP shaft. One proximity probe (keyphasor) is used to measure the rotating speed and rotating phase angle of the RCP shaft.
Development of Integrated NIMS (I-NIMS)

A new version of the NIMS, called I-NIMS (Integrated NIMS) was developed to enhance and substitute the conventional NIMS in 2008 and has been continuously upgraded by Korea Atomic Energy Research Institute (KAERI). The I-NIMS is capable of performing online condition monitoring and integrated structural health diagnosis of the pressure boundary components in NSSS.

The major distinctive feature of the I-NIMS is to make full use of the whole signals obtained from the different subsystems’ field sensors, incorporating the high speed multi-channel signal processing hardware and the data fusion technology. The unfiltered raw signals directly sampled from field sensors such as accelerometers, AE sensors, proximity probes, and ex-core neutron detectors have a common feature in that they can be multi-dimensionally correlated by the phenomena due to the vibration or wave propagation of the reactor pressure boundary components. For example, the impact signal generated by a loose part can be detected not only by the accelerometers of LPMS, but also by the AE sensors of ALMS. Since the sensor locations of the LPMS are different from those of the ALMS, the ALMS sensors can be utilized as virtual sensors for loose part monitoring and diagnosis.

The second outstanding feature of the I-NIMS is that it has employed advanced digital signal processing techniques that can remarkably improve the monitoring and diagnostic capabilities. For example, the new LPMS is equipped with a two dimensional time-frequency analysis (smoothed Wigner-Ville distribution) technique that makes it possible to more accurately estimate the TOADS (Time-Of-Arrival Differences) between different flexural wave groups than ever. In addition, the I-NIMS is including state-of-the-art technical tools such as three dimensional computed tomography algorithm for source localization, a new mass estimation algorithm, the mode separation algorithm for the reactor internal vibration, principal order analysis, and directional spectra, etc.

Summary

The Integrated NIMS (I-NIMS) has been developed and being upgraded by KAERI. The various functional capabilities of the I-NIMS hardware and software have been tested and verified through a site acceptance test performed in Hanbit nuclear power plant unit 3. It is expected that the I-NIMS can enhance the monitoring and diagnostic reliability compared to the conventional NSSS integrity monitoring system.

In order to confirm the capability of the developed I-NIMS, a SAT (Site Acceptance Test) was performed at Henbit nuclear power plant unit 3. The functional capabilities of the hardware and software have been fully tested, verified, and the related new techniques have been registered for patents. Therefore, it is expected that the I-NIMS can exclusively improve the monitoring and diagnostic reliability compared to the conventional NSSS integrity monitoring system.

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Introduction

The reactor pressure vessels are usually constructed by welding large rolled plates, forged sections or nozzle pipes together. These welds should be periodically inspected using sensors such as ultrasonic transducers or visual cameras to assure the integrity of the vessel. Such inspections are usually conducted underwater to minimize an exposure to the radioactively contaminated vessel walls. The reactor vessel inspection has been performed using a conventional inspection machine with a big column. This machine, however, is so huge and heavy that its maintenance and handling is extremely difficult.

In order to resolve these problems, KAERI has developed an underwater mobile robot, which is guided by a laser pointing device, and has performed a series of experiments both in the mockup and in the real reactor vessel. The system is so small and compact that it will reduce the critical path process for a pre-service inspection of pressurized water reactors. By deploying two robots simultaneously in the vessel, the overall inspection time can be greatly reduced. This article describes the outline of the robotic inspection system developed in KAERI’s laboratory.

System Configuration

Our reactor inspection system, called RISYS, consists of a reactor inspection robot, a laser pointing device, a main control computer, a sonic data acquisition and analysis system.

The Reactor Inspection Robot (RIROB) is a submarine type mobile robot whose weight is approximately 50 kg in air and becomes zero in water by the aid of floats. Most of the reactor pressure vessel in a PWR is composed of carbon steel and lined inside with austenitic stainless steel. In order to climb the vertical wall of the vessel, the RIROB has four magnetic wheels: two are caster wheels and the other two are driven by DC servo motors so that the robot can move in any direction on the vertical inner wall of the reactor vessel. The robot can control its linear velocity and angular velocity by the sum and difference of the velocities of the left and right driving wheels. Both the front and rear caster wheels are mounted on the parallelogram links with the robot body plate, so that the robot body can always be parallel to the wall, even though the wall is cylindrical.

The robot also has a light and long manipulator, and the ultrasonic probes are attached to its end effector. The manipulator has five degrees of freedom which are slide forward, twist, rotation, telescopic stretch, and a probe rotation. The manipulator can reach up to 120 cm using 4 consecutive translation links. The camera and lamp are mounted on the robot and the visual image from the camera is transmitted to the main control station. The robot has an inclinometer to measure the inclination of the mobile robot and to control the robot posture. The depth sensor is also mounted on the robot body to measure the water pressure and to calculate the current vertical depth of the robot.

A conventional inspection machine with a huge manipulator can easily place its end effector equipped with an ultrasonic probe to the desired weld position. However, the inspection system using an underwater mobile robot guided by a laser pointing device, needs a lot of calculations and geometric analysis. In order to inspect the welds accurately, the robot should move exactly to the given position.

The laser pointing device induces the robot to the next position. The laser pointing device is fixed in the middle of the crossbeam across the reactor upper flange. The laser pointing device emits a laser beam to the next position for the robot to move to. The robot, with the position sensitive detector (PSD) on its back, detects the deviation of the laser beam spot from the center of the position sensitive detector, and moves in the appropriate direction to make this deviation zero. The laser pointing device is a kind of pan-tilt device on which the diode laser is mounted. The device is accurately driven by the servo motors of which the resolution is less than 0.01 deg/step.
When the laser beam points to a specific position on the PSD surface, the sensor generates currents corresponding to the deviation of the laser spot with respect to the center of the PSD. We control the robot in such a way that the deviations of the laser spot become equal to zero. More complex calculations are needed to inspect the welds near a nozzle using a mobile robot than the other welds in a reactor vessel. When inspecting the welds around a nozzle, the robot moves around this nozzle.

The function of the main control computer is to control the RIROB, the laser pointing device and the sonic data acquisition subsystem. It is a PC based control station with operating software and interfaces. It has the geometric information of all reactor vessels operating in Korea, so that inspections can be planned and simulated on a 3D graphic display.

During inspection, the main control system generates the scan path for the RIROB to move along. Simultaneously, the current posture of the robot is displayed graphically and the image captured by the camera on the robot is also displayed. After inspection, examination reports are generated using the stored data. The system can also be operated in a manual mode during a computer control malfunction.

The computer software includes other convenient modules: an input module of reactor specifications, an inspection procedures module, a selection of inspection items, an automatic location of the inspection robot, previous simulations of the robot movements, a display of the inspection status, communication with the RIROB, laser pointing device and the sonic data acquisition subsystem and a fully automatic inspection and manual inspection. The data acquisition subsystem drives the ultrasonic sensor, and collects, displays and stores the reflected signal data.

Robotic Inspection

In order to confirm the integrity of our developed inspection system, we have performed a series of experiments in the (Continued on page 42)
reactor vessel mockup as well as in the real reactor vessel at the Ulchine nuclear power plant in Korea. Prior to the underwater experiments, we performed the experiment in air to confirm our laser guidance control method. Positioning accuracy is examined, including the rotation and inclination accuracy of the underwater mobile robot, scan path accuracy of the manipulator, angle measurement accuracy of the pan-tilt unit of the laser pointer, and the overall positioning accuracy of this system. The above results of the functional tests indicate that the prototype inspection system satisfied most of the acceptance criteria.

In order to determine whether the weld has defects or not, we also have conducted the ultrasonic testing. After emitting an ultrasonic wave to the suspected welds, we monitored its reflected signal. Usually a reactor pressure vessel is manufactured by welding several parts together. The welds to be inspected in the vessel are largely classified as nozzle welds and circumferential welds. Circumferential welds include the welds of the flange to upper shell, the upper shell to the middle shell, the middle shell to the lower shell, and the lower shell to the bottom head, while the nozzle welds include the welds of the nozzle to the middle shell, and the nozzle to the nozzle pipe so called the safe end. Besides, a flange ligament should be inspected.

When inspecting each weld, we have to use various incident angles of the ultrasonic wave for a more accurate and stricter inspection. For example, in the case of a reactor shell welds inspection, we used incident angles of 0, 45, 60, 50/70 degrees, respectively. In addition, for each incident angle, we have to scan the welds in four directions: upward, downward, and in a clockwise and counter clockwise direction by using ultrasonic probes with a specified incident angle. Thus we have to inspect the welds seventy seven times in total.

A Korean standard reactor vessel has six nozzles, thus the number of nozzle inspections becomes sixty, and the number of circumferential weld inspections becomes sixteen. The probe assembly should be cleverly designed to contact the probe plate to the weld surfaces with a suitable compliance by tilting adjusters and springs.

**Summary**

In order to improve the reactor vessel inspection system, we have developed an innovative robotic inspection system based on underwater mobile robot. We completed the laser induced control of the mobile robot, and the method is thought to be applicable to other industries. Ultrasonic testing for Ulchine 6 reactor vessel was successfully completed at Doosan Heavy Industries and Construction Co. When our system is used for a reactor vessel inspection instead of the conventional machines, a lot of benefits are expected to be the result, such as a critical path process reduction, handling safety improvement, examination reliability and positioning accuracy, and so on.

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Dong-Young Lee
Dong-Young Lee received the B.S. and M.S. degree in Electrical Engineering from the Kyungpook National University in 1984 and 1987 respectively, and Ph.D. degree from the Chungnam National University in 2005. He has been a researcher of the Korea Atomic Energy Research Institute since 1987. His research interests include instrumentation and control system, reliability analysis, and safety assessment.

Summary
Korea developed a Programmable Logic Controller (PLC) for a digital safety system for the APR1400. With the safety-grade platform PLC, a reactor protection system, an engineered safety feature-component control system, and a reactor core protection system were developed, and applied to Shin-Hanul units 1&2.

POSAFE-Q PLC
The safety-grade PLC platform named POSAFE-Q was developed so that it meets the requirements of the Class 1E, 10 CFR 50 Appendix B, and Seismic Category I.

The PLC consists of various modules such as a power module, a processor module, communication modules, digital input/output modules, analog input/output modules, a local bus extension module, and a high-speed pulse counter module. The PLC installs two independent power modules in a rack. The power module has a 100% power supply capability for each. Accordingly, even when there is a fault in one power module, it does not affect the PLC operation. The PLC can extend the number of input/output modules through a local bus extension module. The communication modules consist of Profibus-Fieldbus Message Specification (FMS), High Reliability-Safety Data Link (HR-SDL), and High Reliability-Safety Data Network (HR-SDN).

The processor module uses a Texas Instrument Digital Signal Processing (DSP) CPU and the real-time operating system named pCOS was developed based on the Micro-C real-time operating system. All communication modules were developed based on the Fieldbus protocol to meet the deterministic requirements.

The engineering tool named pSET was also developed. The developers of the application programs can perform programming, debugging, and simulation on the pSET environment. The application program developed on the pSET is downloaded into the processor module through RS-232C. The pSET operates on Windows 2000/NT, which meets the IEC 61131-3 requirements.

Equipment Qualification (EQ) including environmental qualification, seismic qualification and EMI/EMC qualification have been performed according to EQ code and standards. Total man-machine interface system including safety-grade PLC based digital reactor protection system is determined by Korean utility and vendor organizations to apply to Shin-Hanul units 1&2, with a commercial operation date of 2017 and 2018 respectively.

The digital safety system is the plant protection system including Reactor Protection System (RPS), Engineered
The RPS generates a reactor trip signal and the engineered safety feature actuation signals automatically whenever the monitored processes reach the predefined set points. The RPS is designed with the redundant 4-channel architecture, and every channel is implemented with the same architecture. A single channel consists of the redundant Bi-stable Processor (BP), the redundant Coincidence Processor (CP), an Automatic Test & Interface Processor (ATIP), and a Cabinet Operator Module (COM). The BP module generates a logic-level trip signal by continuously comparing the sensor inputs with the predefined trip set-points. The logic-level trip signals generated in the BP module of any channel are transferred to the CP modules of all the channels via the HR-SDL. The CP module monitors the logic-level trip signals transferred from the four BP modules. When two or more logic-level trip signals from the BP channels are activated, the CP modules will activate the output signal for the reactor trip. The ATIP module monitors the operation status of the RPS, and conducts a surveillance test to ensure a reliable operation of the BP and the CP module in the same channel. The test results of the ATIP are transferred to the COM module which has an operator interface facility implemented with an industrial PC and a flat panel display. The BP, CP, and ATIP modules of the RPS are implemented with the PLC platform.

Engineered Safety Feature-Component Control System

It initiates several emergency actuations to prevent the plant from a hazardous state during/after accidents. The actuations include a safety injection, a containment isolation, a main steam line isolation, an auxiliary feedwater injection, and a containment spray actuation.

The ESF-CCS is designed with four redundant divisions (i.e., A, B, C, and D), and implemented with the PLC platform. The principal components of an individual division are fault tolerant Group Controllers (GC), Loop Controllers (LC), an ESF-CCS Test and Interface Processor (ETIP), a Cabinet Operator Module (COM) and a Control Channel Gateway (CCG).

Each GC receives ESF initiation signals from the RPS and radiation monitoring system via fiber-optic receivers. All GCs perform system level Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) ESFAS logics independently, so that they can transfer the system level ESF actuation signals to all LCs in the division. LCs perform a component control logic using system level ESF actuation signals from GCs or component level control signals from an operator, so that the output control signals are assigned to individual plant components. ETIP takes charge of the passive and active test functions of the ESF-CCS and the interfaces of the ESF-CCS with other systems such as the RPS and qualified indication and alarm system. COM provides information about an ESF actuation status, an ESF component status and a module status. CCG supports the interface between the ESF-CCS and soft controllers in the main control room or remote shutdown room.

Reactor Core Protection System

The core protection calculator system provides an on-line calculation of

(Continued on page 46)
the Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) of the plant with an improved algorithm and a different system configuration compared to the existing system. The improved algorithms of RCOPS include DNBR algorithm improvement for the core thermal margin, resolution of the latching problem of the false Control Element Assembly (CEA) signal, and addition of a pre-trip alarm generation. It generates a reactor trip signal when the condition exceeds the DNBR or LPD design limit. It consists of four independent channels employing a two-out-of-four trip logic.

Each channel of RCOPS consists of a Core Protection Processor (COPP), a Control Element Assembly Processor (CEAP), and two Channel Communication Processors (CCP). COPP monitors the CEA positions of one quadrant of the reactor core. These CEAs are called the target CEA of that channel. COPP generates planar radial peaking factors that are used to calculate DNBR and LPD. CEAP takes all CEA positions of the reactor core through CCP and examines the CEA deviation between subgroup positions. If this deviation is higher than a specified value, CEAP will send the penalty factor to COPP.

**Software Qualification**

Most of the software embedded in the PLC is classified as safety-critical software. All the software and firmware of the POSAFE-Q PLC were developed and verified following the software development life cycle Verification and Validation (V&V) procedure. The main activities of the V&V process are the preparation of the software planning documentsations, verification of the Software Requirement Specification (SRS), Software Design Specification (SDS) and codes, and a testing of the software components, the integrated software, and the integrated system. In addition, a software safety analysis and a software configuration management are included in the activities.

The SRS V&V activities consist of a technical evaluation, a licensing suitability review, an inspection and a traceability analysis, a formal verification, preparation for an integrated system test plan, a software safety analysis, and a software configuration management. The code V&V activities include a traceability analysis, a source code inspection, a component test case, a test procedure, and a test report generation, a software safety analysis, and a software configuration management. Testing is the major V&V activity of the software integration and system integration phase. A software safety analysis at the SRS and SDS phases uses the Hazard Operability (HAZOP) method, and at the implementation phase the source code has been evaluated using the safety programming guide of NUREG/CR-6463 (Review Guidelines on Software Languages for Use in Nuclear Power Safety Systems). Finally, the software configuration management was performed using the Nuclear Software Configuration Management (Nu-SCM) tool.

**Upgraded PLC**

Since the results of the R&D project applied to the actual nuclear power plant construction, there are lots of additional requirements from the utility for PLC. Some features to improve the functionality of PLC were added. The self-diagnostic, heartbeat, and automatic recovery functions have added for the HR-SDN module of the PLC. The feature such as self-diagnostic in HR-SDL is added for the HR-SDL, and the piggy back in the form of a single module has been changed. The input latch and fail-safe features were added for some input and output modules. The dual processor module and a pSET-II as a new engineering tool were developed.

**Conclusions**

The POSAFE-Q PLC as a safety-grade platform in digital safety system was developed and has obtained the licensing approval from Korean regulatory body via a topical report. The safety evaluation reports were issued in February 2009. Furthermore, a Korean utility and vendor company determined to apply safety-grade PLC to the Shin-Hanul units 1&2 nuclear power plant in March 2009.

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Sanmen Nuclear Plant

In 2007, Westinghouse Electric Company LLC was awarded a contract to build four units of AP1000® nuclear power plants in China: two in Sanmen and two in Haiyang. About two years later, the first pour of concrete at Sanmen was completed on April 19, 2009. This important milestone marked the beginning of the Sanmen AP1000 nuclear power plant – the first AP1000 ever to be built was underway.

Sanmen Nuclear Power Project (NPP) is located in Sanmen County, Taizhou City, Zhejiang Province – an eastern coastal area of China. It is 171 km north of Hangzhou City, 83 km east of Ningbo City, 51 km west of Taizhou City and 150 km south of Wenzhou City.

Sanmen Nuclear Power Company, Ltd. (SMNPC) was established on April 17, 2005 and falls under the management of China National Nuclear Power Co. Ltd. (CNNC) – the largest nuclear power utility company in China. With 51 percent of the shares of the project’s total investment, CNNC is in charge of a full spectrum of the power plant’s work, including construction, commissioning, operation and management.

By adopting AP1000 technology, the latest Generation III+ pressurized water reactor (PWR) nuclear power technology from Westinghouse, SMNPC will build six units of nuclear reactors in three phases with 1250 megawatt electricity (MWe) each, and the total installed capacity expected to reach 7500MWe.

This is China’s first indigenous program, approved by the State Council on July 21, 2004. Sanmen Nuclear Power Project is viewed as the biggest energy cooperative project between China and the U.S., and it is also the largest investment project in Zhejiang Province.

The first Sanmen unit is expected to come on-line by the end of 2015. In 2015, the electricity generation from the Sanmen Nuclear Power Project will account for 3.3 percent of the total electricity generation of Zhejiang Province.

During the course of the construction of the Sanmen plants, China is enhancing its capability in nuclear construction, engineering and manufacturing, while the country also develops its own Generation III technology.

Safety of the Site

Anti-tsunami geological conditions

The Sanmen site is located on the wide and gentle continental shelf with no big geological fault zone around the plant. The water within 100 km of the site is about 0-100m deep.

There has been no record of earthquake or tsunami in the coastal area of the site since A.D. 708. The island chain from the Ryukyu Islands to Taiwan Island, the Okinawa ocean trench and the continental shelf constitute a natural protective barrier against tsunami for the site along Zhejiang coast.

Seismic Design

The actual safe earthquake of the utmost limit (SL-2) for the site is 0.15g,
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Multiple Defense... (Continued from page 48)

while the safe shutdown earthquake (SSE) adopted in the design is 0.3g, which leaves a big safety margin.

Safety features:

The AP1000 nuclear power plant, designed by Westinghouse, is a two-loop pressurized water reactor (PWR) that uses a simplified, innovative and effective approach to safety. With a gross power rating of 3,415 megawatt thermal (MWt) and a nominal net electrical output of 1,250 MWe, the AP1000 reactor, with a 157-fuel-assembly core, is ideal for new baseload generation.

The AP1000 PWR is based on a simple concept: in the event of a design-basis accident, such as a main coolant-pipe break, the plant is designed to achieve and maintain safe shutdown condition without operator action, and without the need for AC power or pumps. Rather than relying on active components, such as diesel generators and pumps, the AP1000 plant relies on natural forces - gravity, natural circulation and compressed gases - to keep the core and the containment from overheating.

The AP1000 PWR provides multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core-damage probabilities while minimizing the occurrences of containment flooding, pressurization and heat-up. Defense-in-depth is integral to the AP1000 plant design, with a multitude of individual plant features including the selection of appropriate materials; quality assurance during design and construction; well-trained operators; and an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits. The AP1000 PWR is also the only Generation III+ reactor to receive Design Certification from the U.S. Nuclear Regulatory Commission (NRC).

Based on nearly 20 years of research and development, the AP1000 plant builds and improves upon the established technology of major components used in current Westinghouse-designed plants. Components such as steam generators, digital instrumentation and controls, fuel, pressurizers and reactor vessels are currently in use around the world and have years of proven, reliable operating experience.

Environment protection

After the completion of Sanmen Phase I NPP, compared with ultrasupercritical coal-fired units of the same capability, the two units may help reduce transportation of five million tons of high-quality coal, emission of 11,490 tons of sulfur dioxide, 19,088 tons of nitrogen oxide and 1,345 tons of smoke and dust.

Contact: Jackie Smith, Westinghouse Electric Company, 1000 Westinghouse Dr., Cranberry Township, Pennsylvania 15632; telephone: (412) 374-3372, email: smith1jp@westinghouse.com.

Internal view of Sanmen Unit 1 as of the end of March 2014. Photo courtesy of the Sanmen Nuclear Power Project.
Westinghouse provides comprehensive, integrated services and solutions to the decommissioning and dismantling (D&D) and waste management industries. We have extensive experience in the dismantling of nuclear installations, from uranium mill plants to nuclear power plants. We provide state-of-the-art solutions for spent fuel services and for the treatment and handling of radioactive waste. Westinghouse offers proven solutions for the interim storage and final disposal of low-, intermediate- and high-level waste.

Our dedication to a cleaner environment extends to servicing existing nuclear power plants and managing by-products in an environmentally responsible manner.

For more information, visit us at www.westinghouse.nuclear.com.
Adding a Supplemental Diesel Generator may address loss of AC Power due to "Beyond Design Basis" events, provide more plant flexibility, and improve electrical power reliability in both normal and abnormal conditions.

Zachry offers extensive capabilities and experience to ensure the successful integration of Supplemental Diesel Generators. These units have their own controls, fuel supply, and switchgear connections to Safety Buses. Our experienced and knowledgeable engineers, designers, and constructors are ready to work with you to provide cost-effective, comprehensive and quality solutions including:

- Options analysis
- Equipment specifications
- Conceptual/detailed modification packages, including drawings, analysis, and calculations
- Field engineering, construction, startup, and closeout support

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