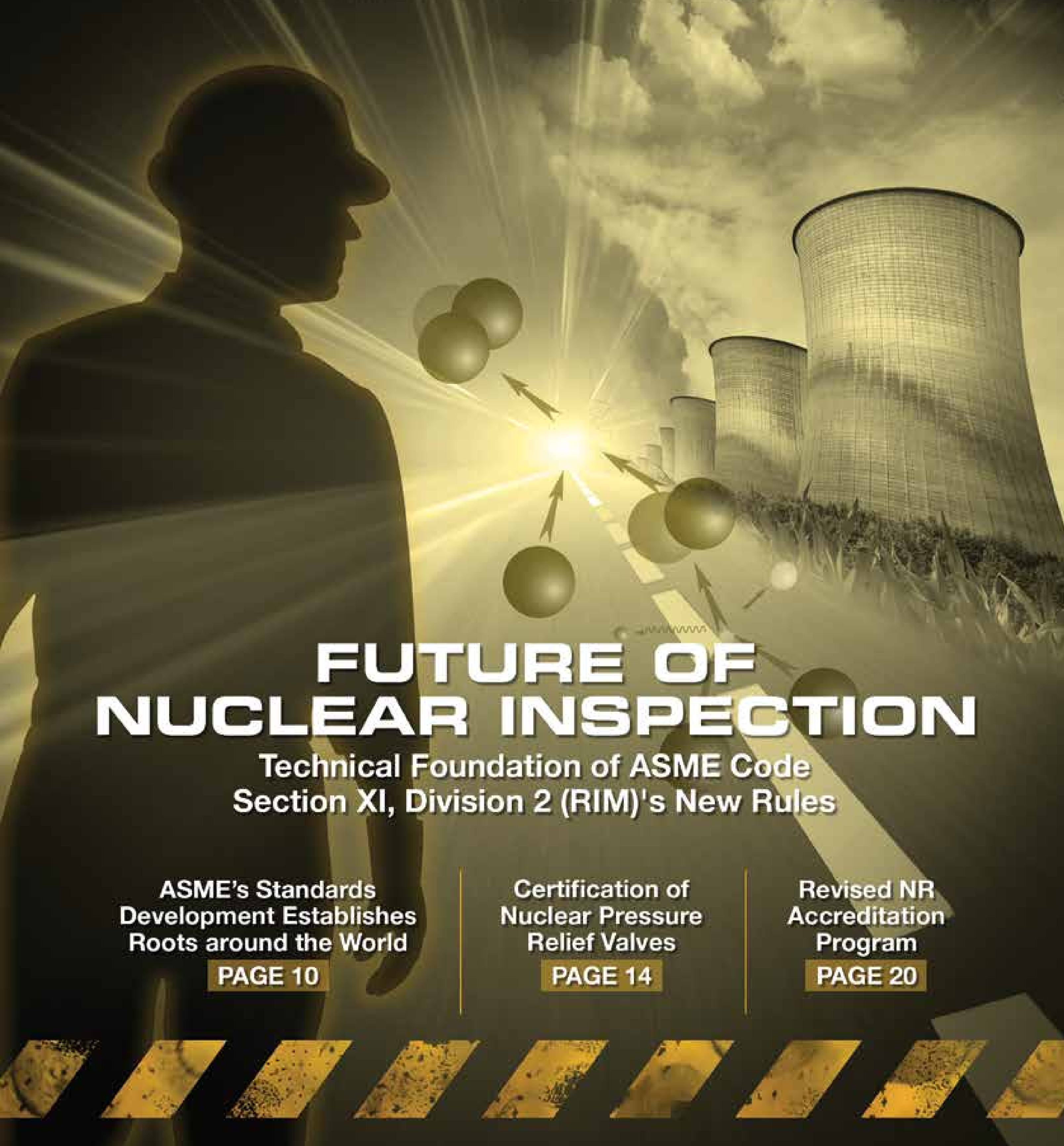
SUMMER 2015

BULLETIN

TECHNICAL JOURNAL OF THE NATIONAL BOARD OF BOILER AND PRESSURE VESSEL INSPECTORS



FUTURE OF NUCLEAR INSPECTION

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BULLETIN

The National Board of Boiler and Pressure Vessel Inspectors was organized for the purpose of promoting greater safety by securing concerted action and maintaining uniformity in the construction, installation, inspection, and repair of boilers and other pressure vessels and their appurtenances, thereby ensuring acceptance and interchangeability among jurisdictional authorities empowered to ensure adherence to code construction and repair of boilers and pressure vessels.

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Godspeed, Mr. Justin

BY DAVID A. DOUIN, EXECUTIVE DIRECTOR

Al Justin enjoyed the distinction of being the only National Board Executive Director to retire.

And that was just like Al: one of a kind.

His passing several weeks ago [see page 34] brought back a flood of memories, many good and others reflecting a time when uncertainty encircled the National Board. Misdirected assertions; miscommunications; and misunderstandings amongst members, the Board of Trustees and its executive director – and finally the death of the sitting executive director – came together to create an atmosphere that would prompt a new era for the organization.

Al took over during some pretty rocky times for the National Board. But he was the right guy to implement new policies that would set the organization on a proper course.

And so, in March of 1993, former Minnesota Chief Boiler Inspector and past National Board Board of Trustees Chairman Albert Justin was elected by membership as the organization's fifth executive director. No stranger to the issues besetting the National Board, Mr. Justin hired new administrators to help change the professional culture.

In his very first Executive Director's Message for the spring 1993 *BULLETIN*, Mr. Justin declared he would "focus on National Board's professional relationships and the priority thereof."

Mr. Justin's mission was twofold: organization transparency and getting people working together. Through a series of new institutional policies, National Board members and staff were made aware of what they could expect and what would be expected of them. Efforts were launched to bring the National Board closer to other standards-developing organizations, safety and engineering groups, manufacturers, insurance companies, and regulators. He encouraged the National Board to be more responsive to every organization and association sharing the public's trust.

For the next eight years before his retirement in 2001, Mr. Justin oversaw revision of the National Board Commission Examination reflecting new education requirements; administration of the first National Board Commission Exam outside of North America; adoption of new rules for commissioned inspectors; and development of new rules to evaluate effects of rupture disk devices on pressure relief system capacity. He literally changed the landscape of National Board training through construction of the National Board Training and Conference Center. Equally significant, he coordinated National Board's entry into electronic communications, including the Internet and Electronic Data Transfer.

The years under Mr. Justin's leadership were among the most energized, if not challenging, periods in the National Board's long and distinguished history. And while his contributions to our organization were numerous, his efforts on behalf of our industry strengthened it to become the successful entity it is today.

In a farewell interview in 2001, Al was asked by the *BULLETIN* if there was anything he failed to achieve at the National Board. He responded in a style so typical of the fifth executive director:

"If anything went unaccomplished, it was only because I ran out of time."

As do we all.

Albert J. Justin was 88 years old. ♣

National Board Presented Chinese Translation of the *National Board Inspection Code*

BY GARY SCRIBNER, MANAGER OF TECHNICAL SERVICES

Representatives from The National Board of Boiler and Pressure Vessel Inspectors were presented with the new Chinese translation of the *National Board Inspection Code* at the 2014 National Special Equipment and Energy-Savings Science & Technology Week in the city of Dongguan, Guangdong province, China, November 19 to 25, 2014.

The event was jointly sponsored by The General Administration of Quality Supervision, Inspection and Quarantine (AQSIQ) and the China Special Equipment Inspection and Research Institute (CSEI), the two main groups responsible for public safety in China, and was attended by more than 500 foreign and domestic technical experts and students.

National Board's John Burpee (chairman of the Board of Trustees), Charles Withers (assistant executive director – technical), and this author were presented with the new Chinese translation of the *National Board Inspection Code* (NBIC) by CSEI delegates. The translation process began in 2012 when the National Board authorized CSEI to translate the NBIC (2011 version) into Chinese. The translation and publishing process took more than two years to complete and was finalized in September 2014. Both the National Board and CSEI hope that the Chinese translation of the NBIC will strengthen bilateral communications between the two groups and raise the level of international boiler and pressure vessel knowledge and public safety.

National Board and China Special Equipment Inspection representatives present Chinese NBIC Translation.

The theme for the event was "Past, Present, and Future." The opening session presentation included an overview of the history of China with a focus on how the country's technological evolution has been affected by war (including two world wars) and lack of acceptance of scholars in the social classes well into the 1970s. This presentation concluded with a recap of how the acceptance of scholars has led to the vast technological revolution that China has benefited from over the last 30 to 40 years.

In keeping with the theme of the event, Charles Withers and then-American Society of Mechanical Engineers (ASME) Deputy Executive Director June Ling both gave presentations overviewing the past, present, and future of their respective organizations.

AQSIQ is a ministerial administrative organization directly under the State Council of the People's Republic of China in charge of national quality, metrology, entry-exit commodity inspection, entry-exit health quarantine, entry-exit animal and plant quarantine, import-export food safety, certification and accreditation, standardization, and administrative law-enforcement. CSEI is a semi-governmental organization and the sole national technical organization responsible for the inspection and research and development of special equipment in China.

The ongoing communication between the National Board, ASME, and CSEI continues to strengthen public safety initiatives on an international level. The National Board looks forward to future meetings and thanks AQSIQ and CSEI for their hospitality and shared commitment to pressure equipment safety. ✤

FUTURE SHIFTS IN NUCLEAR INSERVICE INSPECTION

How ASME Section XI, Division 2 (Reliability and Integrity Management) is Developing New Rules to Accommodate Advanced Reactor Designs and What it Means for the Future of Inspection

By A. Thomas Roberts, MPR Associates Inc.

In the last issue of the National Board *BULLETIN*, the article "An Overview of Small Modular Reactors (SMRs)" identified a need for changes to current *ASME Boiler and Pressure Vessel Code* (ASME Code), Section XI, Division 1, *Rules for Inservice Inspection of Nuclear Power Plant Components.* New rules are needed to accommodate innovative SMR designs and other non-SMR advanced design reactors to ensure long-term reliability and safe operation of these specific designs. The new rules will require a shift from the inservice inspection activities presently employed in ASME Code Section XI, Division 1, and used by inspection personnel.

The ASME Code Section XI Standards Committee is developing ASME Code Section XI, Division 2, entitled *Reliability and Integrity Management* (RIM) to address these advanced reactor designs. ASME Code Section XI, Division 2, will be a "technology neutral" inservice code that may be applied to all advanced reactor designs, including SMRs. Included in Division 2 will be technology-specific appendices that are intended to account for different reactor designs with regard to inservice inspection (ISI) parameters.

In order to understand how authorized nuclear inservice inspector (ANII) inspections may change in the future, an overview of the technical basis for Division 2 and the process it employs is provided herewith.

The foundations of ASME Codes Section XI, Division 1; and Section XI, Division 2, are fundamentally different. In order to understand these differences, it is important to acknowledge the original technical foundation for ASME XI, Division 1. The explanation is summarized well by a quote from one of the founding chairmen of ASME XI, Division 1:

*"The philosophy of Section XI is to mandate a sufficient number of examinations and tests (**selected deterministically**) to provide assurance that the original safety that was designed and built into the plant is maintained throughout its service life." – L.J. Chockie (1975) - Chair, Section XI.*

In contrast to ASME Code Section XI, Division 1, which was founded on deterministic criteria, ASME Code Section XI, Division 2, is built on a System Based Code (SBC) technical approach. This approach was chosen because it evaluates all systems and components for their relative consequences for maintaining overall plant safety and then establishes appropriate monitoring parameters to ensure long-term reliability.

This is opposed to the prescriptive approach used by Division 1, which uses the Class 1 (e.g., reactor coolant system), Class 2 (emergency core cooling systems), and Class 3 (e.g., tertiary systems) approach to ISI with each Class having less-rigorous criteria for ongoing monitoring.

As an introduction for the reader, SBCs provide a framework to permit increased flexibility to:

- Give a rational method for safety margin optimization in order to increase in-service monitoring and operational maintenance flexibility.

  Note: *Safety margin* is a term describing the structural capacity of a system beyond the expected loads or actual loads. Essentially, how much stronger the system is than it usually needs to be for an intended load(s).

  Researchers Shigeru Takaya and colleagues explained one of the key concepts of safety margin optimization is providing "a framework that intends to allow the optimum allocation of margins on the structural integrity of components encompassing various technical aspects in a plant life cycle, such as material, design, fabrication, installation, inspection, and repair and replacement. By fully taking account of these technical characteristics, the SBC concept pursues improved reliability and economy while meeting the plant safety goals." (1)

- Enable optimization of safety margin integrity for the entire life cycle of systems, structures, and components (SSCs).

- Continuously evaluate safety margin integrity from initial design through plant decommissioning as a "*living*" program.

A simplified description of the process employed by SBCs, and employed by ASME Code Section XI, Division 2, RIM, follows:

1. Determine scope of SSCs for the RIM Program.

2. Evaluate SSC damage mechanisms.

3. Determine plant- and SSC-level reliability requirements.

4. Evaluate RIM strategies to achieve reliability targets.

5. Evaluate uncertainties in SSC reliability performance.

6. Determine scope and parameters for the specific SSC in the RIM Program.

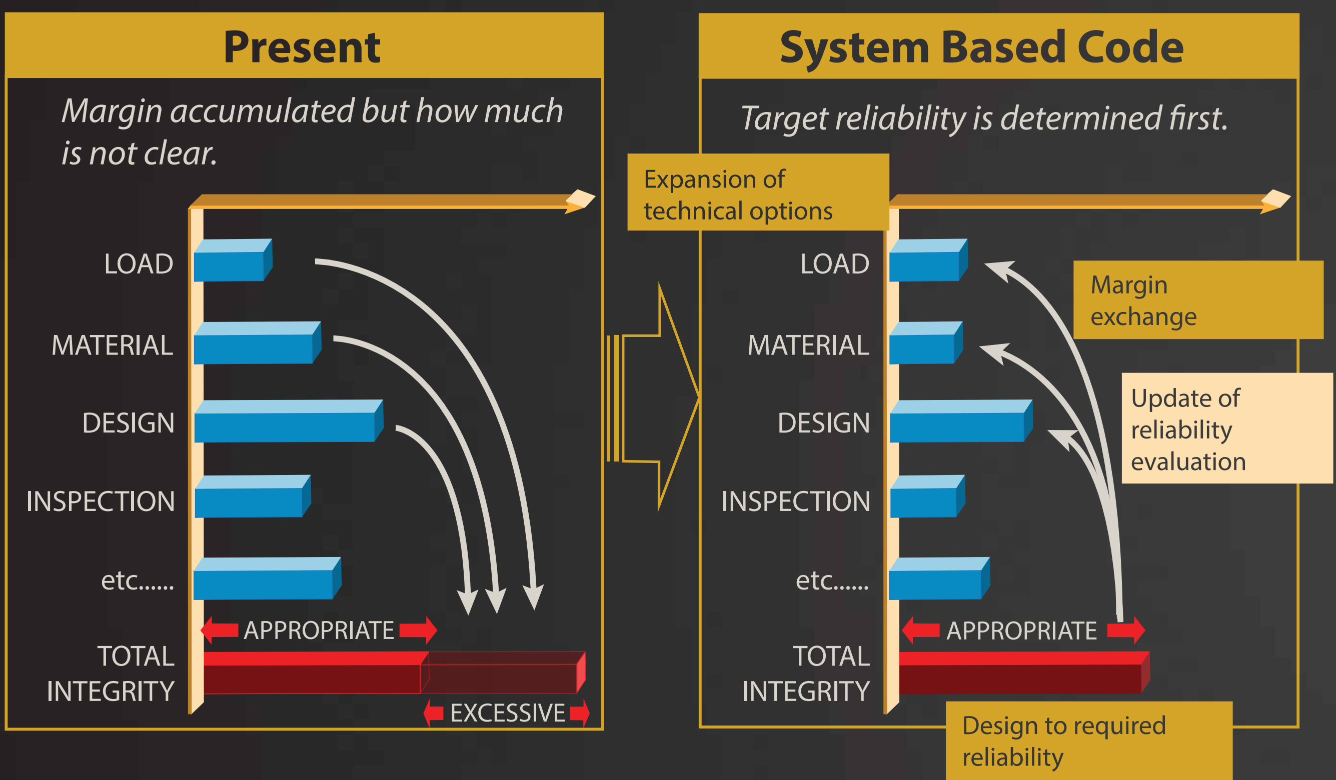
7. Continuously monitor SSC reliability performance and update the revisions to a RIM Program over the entire life of a reactor facility.

In a SBC, a total required safety margin is first defined and then it is distributed to each individual requirement in a rational manner. As noted previously, this approach contrasts with the conventional construction and inservice codes, which are based on deterministic approaches.

A comparison of the two different approaches is illustrated in Figure 1.[2]

Figure 1

Tai Asayama and colleagues explained the SBC concept consists of three parts: 1) design to target reliability that must be met throughout the service life, 2) margin exchange among the various technical areas of concern such as design, inspection, fabrication, and fitness for service, and 3) expand technical options by the timely adoption of newly developed technologies that are not in current codes and standards. Schematic illustration of the concept is given in Fig. 1. [Asayama, et al., Reference 2.]



Inservice Inspection (ISI) Requirements Based on System Based Code (SBC) Concept

Each of the seven steps outlined below provides a high-level overview of the SBC process and summarizes various evaluations involved, each which fulfill different objectives. The information was drawn from and is an overview of the detailed analyses reflected in References 1 and 2.

1. At the onset, a structural design-oriented evaluation considers SSCs' structural integrity; in other words, the probability of failure based on design conditions.

2. After SSCs are identified for the RIM process, all potential degradation mechanisms (DMs) are considered, including those that are not explicitly addressed in design and construction codes.

The concept of a DM assessment is to consider whether any of the following DMs might apply to specific reactor designs and related components:

- Design characteristics, including materials, component type, and other attributes related to the system configuration.
- Fabrication practice-introduced DMs, including welding and heat treatment.
- DMs introduced by operating and transient conditions, including: temperatures, pressures, and gas/water flow, fluid quality (e.g., primary water, raw water, dry steam, chemistry control, etc.), and other service environments (e.g., humidity, radiation, etc.).
- DMs based on plant-specific or industry-based service experience, if available.
- Results from pre-service, inservice, and augmented examinations and the presence and impact of any prior repairs in the SSC.
- Manufacturer's recommendations for examination, maintenance, repair, and replacement.

These DMs can then be modeled and evaluated to the extent that otherwise is not considered in most design codes. For example, crack propagation and resultant fracture could be included in an evaluation if appropriate.

3. Once the DM assessment is completed for an SSC, all credible failure modes are identified based on the associated degradation mechanisms for each SSC being evaluated.
4. When the DM assessment is completed, the probability of each failure mode is assessed for the SSC.
5. At this stage of the process, the SBC evaluations additionally consider the safety functions of the plant, taking into account events that have been postulated in the safety analysis of the plant and Probabilistic Risk Assessment (PRA), such as:

   - The plant operating state (e.g., mode or operational condition, such as hot standby) relevant to the plant-level risk and reliability goals and SSC-level reliability targets.
   - Initiating events, including internal events and events associated with internal and external plant hazards.
   - Event sequence development sufficient to support the quantification of mechanistic source terms and offsite radiological consequences consistent with applicable regulatory limits on the frequencies and consequences of licensing basis events.

   The main focus of evaluations in the SBC process used by RIM is on the probability of occurrence of an event (e.g., the maximum allowable break from the viewpoint of the plant safety analysis and regulatory criteria). The ability to adequately identify and mitigate an event(s) as defined in a plant's safety analysis (e.g., detect a pipe break) is also considered in this phase of the SBC evaluation.

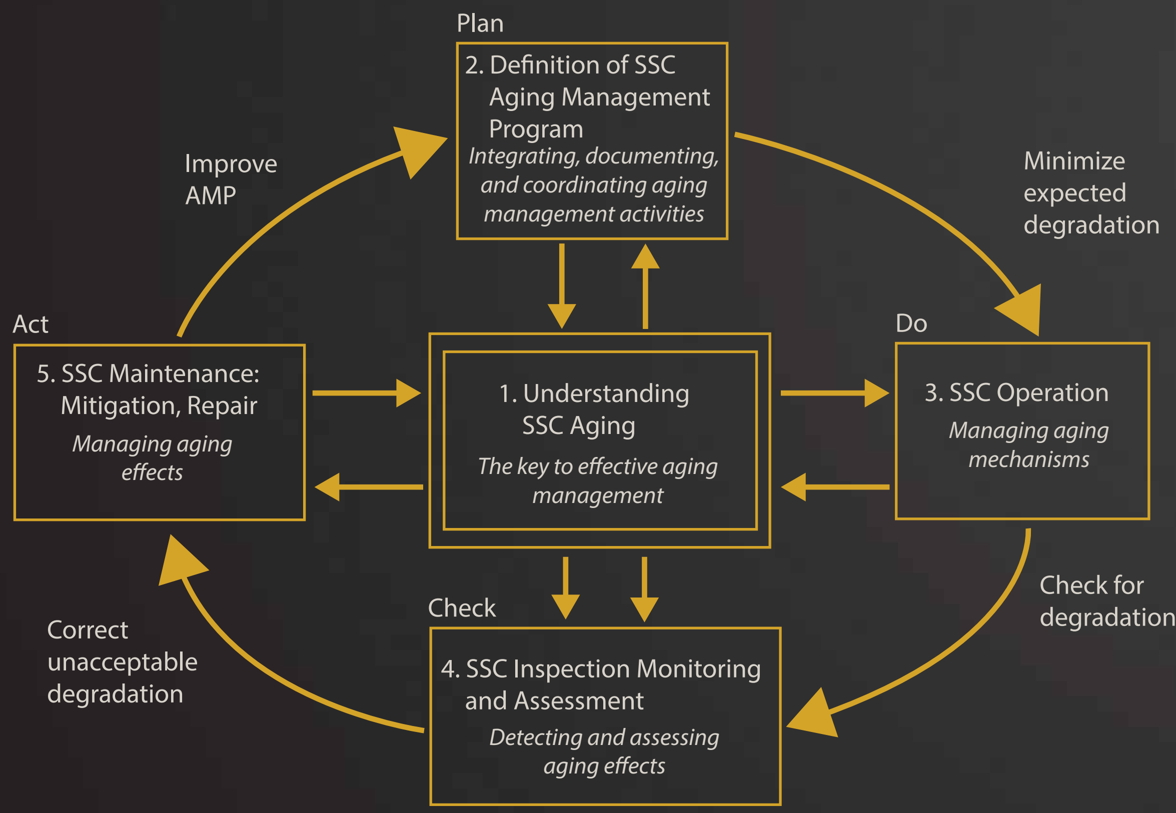
DM assessment, credible failure modes, and established failure mode probabilities are integrated into a plant's PRA to establish SSC's required reliability targets.

6. The aggregated information from all previous evaluations is then used to select meaningful inservice inspection (ISI) methods with practical inspection frequencies that would then be used in developing specific ISI provisions for any given particular SSC.

7. Like most inservice inspection programs that are founded on risk insights, it is expected that once a RIM program is developed and implemented, it would be regularly monitored for effectiveness and employ a continuous feedback loop used to update and make adjustments to the program as additional operating experience is obtained.

This integrated process is the foundation of SBC concepts including ISI applications. It is the overarching foundation for the development of ASME Code Section XI, Division 2 (RIM), and when employed, will serve to effectively provide high reliability for critical SSC – from initial operations through end-of-life service – by effectively creating a comprehensive aging management program (AMP) that protects SSC reliability over the entire life cycle. See Reference 1 for in-depth details about this process. This overall concept is illustrated in Figure 2 below.

Figure 2[3]
Systematic Approach to Aging Management of an SSC



ASME Code Section XI, Division 2, and the National Board

Shifts in an inspector's approach to inservice inspection of advanced reactors are necessary in areas such as ISI cycles and ISI examination methods, and specialized for each reactor design.

Under current ASME Code Section XI, Division 1, rules, ISI examinations are prescriptively required to be performed at discrete time periods during a typical 10-year inservice inspection interval. Some advanced reactor designs may be designed for longer fuel cycles than today's typical PWR or BWR 18-to-24-month fuel cycle. This means that the typical 10-year ISI program interval may not be well-suited for these advanced reactor designs. *Real time* online monitoring of an SSC may provide for greater long-term reliability and assurance of safe operations for a particular SSC.

Further, the use of traditional ISI examination methods such as ultrasonic testing (UT), liquid penetrant (LP) testing, etc., may need to be replaced with more appropriate monitoring or surveillance techniques, such as online acoustic monitoring or periodic surveillance specimen testing (e.g., evaluate the onset of creep damage that may be applicable to some designs).

These examples are departures from the historic approaches to ISI and are important to recognize. In particular, they are essential for authorized inspection agency personnel to internalize, since many inservice inspections or monitoring functions in Division 2 may cause a significant paradigm shift with respect to the day-to-day responsibilities for authorized nuclear inservice inspector (ANII) personnel.

Using ASME Code Section XI, Division 1, generally entails that an ANII monitors compliance with the prescriptive examinations and tests delineated in the various examination tables found in ASME XI, Division 1. In contrast, the role of the ANII in carrying out inspection functions with the RIM process will necessitate that individuals become familiar with the plant-specific design and review the plant-specific RIM program documents. This will be essential to ensure that all of the established inspection methods, surveillance, or monitoring criteria that apply to a specific SSC are completed in accordance with the RIM program criteria relevant to that particular design.

Further, since each unique reactor design will likely be expected to have its own unique RIM parameters, ANIIs who perform inspections at one facility that has one specific design may need to reeducate themselves with different RIM criteria, if their career assigns them to another facility with a different reactor system.

Conclusion

The future of advanced nuclear reactors, with their varied designs, will require a substantial shift in the performance and approach to ISI activities for both the plant operators and authorized inspection agency personnel, and in particular, authorized nuclear inservice inspectors. While the industry is making these advances, including the development of new codes and standards, the familiarization and training of AIA personnel with the RIM process will also be paramount to the future success of any new reactor designs.

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1. Shigeru Takaya et al., "Application of the System Based Code Concept to the Determination of In-Service Inspection Requirements," *Journal of Nuclear Engineering and Radiation Science* 1, No. 1, 011004 (2015).

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3. Proactive Management of Ageing for Nuclear Power Plants, Safety Report Series No. 62, IAEA Vienna (2009).

Photo courtesy of ASME

ASME's Standards Development Establishes Roots around the World

BY CHRISTIAN SANNA, PROJECT ENGINEERING MANAGER, ASME, NUCLEAR S&C ASIA LIAISON, AND RYAN CRANE, PROJECT ENGINEERING MANAGER, ASME, NUCLEAR S&C EUROPE LIAISON

It has long been accepted that the high quality of ASME's codes, standards, and conformity assessment programs is a reflection of the corps of over 5,300 volunteers who participate on over 700 standards development committees. The process is well-served by the extraordinary level of expertise and diversity of perspectives found among our committees' membership. These two qualities lend themselves to the technical excellence and the broad acceptability of ASME products and services, which are used in over 100 countries.

International Outreach

Over the years, ASME Standards and Certification (S&C) has conducted many outreach efforts supported by volunteer leaders, committees, and staff to engage international stakeholders. Meetings, workshops, conferences, and collaborations with our stakeholders and other standards developers are a significant part of ASME's globalization activities, and are vital to our success. As noted by June Ling, retired ASME Deputy Executive Director, keeping our standards and certification products technically and globally relevant is among our most significant objectives.

Recognizing the importance of this objective, ASME Standards and Certification has sought to increase the number and diversity of qualified participants from among our stakeholders. Year-over-year we have been successful; not only has our pool of standards committee volunteers increased, but participation by experts based outside North America has consistently risen as well. This is particularly important as technological development, industry, and commerce are increasingly globalized, and ASME standards increasingly reflect the circumstances of users around the world.

Managing Collaboration, Reducing Burdens

Managing the collaboration of more than 5,300 volunteer experts is one of the most important roles of ASME, and the challenge is compounded as our standards committees become more global with the inclusion of members from around the world. It's clear that facilitating collaboration – and expanding our standards committee membership – requires actively adapting and innovating to reduce the burdens that prevent otherwise qualified experts from participating.

The most significant of these burdens are associated with in-person attendance of committee meetings, which are typically held several times per year and include travel expenses and time away from the office. These factors can be more challenging for volunteer members who must travel from overseas to their committee meetings which, at present, convene predominantly within the US. Nevertheless, a significant number of our volunteer members – currently 17% and rising – reside outside the US. Their impressive efforts over the years to regularly attend meetings are very much appreciated by their committees and ASME. Still, the expectation of attendance is prohibitive for many others.

Innovating for Participants

A number of innovations in our procedures and processes have made participation easier for our volunteers, such as our online standards development management system, CS-Connect, which provides a secure, round-the-clock environment for conducting much of a committee's business outside of face-to-face meetings.

Other improvements include alternative categories of membership, devised specifically to address the challenge of meeting attendance. Participants designated as Delegate, Corresponding, or Contributing Members are not expected to attend meetings. While suitable for a discrete portion of a committee's participants, these membership alternatives have drawbacks: the lack of face-to-face interaction with committee peers, restrictions on some participation privileges, and a practical limit on the number of such members a committee can accommodate.

International Working Groups

A recent innovation avoids these drawbacks and permits many more qualified experts from outside North America to participate in ASME standards development. An International Working Group (IWG) is an ASME standards-writing body composed of a number of members located in a common geographic region outside of North America. Stated plainly, IWGs are ASME standards-writing committees that convene meetings in the country or region where their volunteer members reside.

An ASME Standards Committee – the "top level" group chiefly responsible for establishing consensus and approving all standards actions prior to publication – is usually supported by a number of subordinate groups (i.e., subcommittees, subgroups, working groups, etc.), which are typically delineated by an engineering discipline or technology (e.g., design, high-temperature materials, etc.) and perform much of the development and refinement of proposals for updating our standards. Certainly, membership on ASME standards-writing committees is granted without regard to nationality or residence, and indeed, has many members from across the globe who work together within the traditional committee structure, and ASME appreciates those efforts. However, prior to the implementation of International Working Groups, a design expert from Seoul, for example, could only find a role within ASME's standards-writing activities by meeting the membership expectations of a particular committee – including those for meeting attendance. IWGs enhance our globalization and outreach by facilitating participation by dedicated experts based overseas who are unable to make a significant commitment of time and travel.

International Working Groups pool qualified volunteer members in a subordinate group assembled according to their geographic location. So, in the example presented earlier, the design expert from Seoul may join with other qualified experts as members of a "Korea International Working Group." The Korea IWG supports a particular standards committee and contributes to the improvement of the standard's international applicability and acceptance.

This seemingly simple concept provides profound benefits to our standards-writing participants, standards committees, stakeholders, and ASME, the most obvious being that experts based overseas have enhanced opportunities for participation with greatly reduced impositions of travel. IWGs also offer improved collaboration opportunities with their respective committee colleagues. Standards committees benefit from a broader pool of subject matter experts with valuable perspectives on regional conditions, issues, and applications; some IWGs accommodate as many as 40 expert volunteer members. It is also expected that IWGs will foster the development of our future S&C volunteer leaders hailing from everywhere ASME's standards are used. For local stakeholders, IWGs may facilitate discussion of their experiences and needs, and potentially provide a first line of support and coordination for their inquiries and exchange of technical information.

The first IWGs were established in 2009 by four ASME B31 standards committees. Since then, a large number of IWGs have been founded by our nuclear and boiler and pressure vessel committees. At present, these IWGs are in operation:

India

B31.1 Power Piping
B31.3 Process Piping
B31.4 Liquid and Slurry Transportation Systems
B31.8 Gas Transmission and Distribution Piping Systems
BPV Section I - Construction of Power Boilers
BPV Section III - Construction of Nuclear Power Plant Components

China

BPV Section II - Materials
BPV Section III - Construction of Nuclear Power Plant Components
BPV Section XI - Inservice Inspection of Nuclear Power Plant Components
Operation and Maintenance of Nuclear Power Plants
(At least four additional IWGs are in various stages of development.)

Germany

BPV Section III - Construction of Nuclear Power Plant Components

Italy

BPV Section VIII -Construction of Pressure Vessels

Korea

BPV Section III - Construction of Nuclear Power Plant Components

Europe

Nuclear Quality Assurance (NQA)

Looking Ahead: Increasing Global Participation

Regions outside North America are increasingly significant sources and markets for the equipment and industries served by ASME standards. Through innovation, the knowledge and experience of stakeholders around the world are finding their way into ASME's standards, ensuring continued technical relevance and excellence. Stakeholders interested in participating in any of ASME's standards development activities are encouraged to contact ASME at: www.asme.org/about-asme/get-involved/.

BULLETIN Interview with Ryan Crane and Christian Sanna

BULLETIN: How are IWGs established?

Crane: *The path to each IWG we've established has been somewhat different, but generally there was a recognized benefit for both the standards committee and the volunteer-members to work together on the updates and improvement of our standards. Actually, the motivation to contribute to ASME standards – for the betterment of the standard and improvement of the applicability and user experience – is common to all our volunteers-members, and international engagement with ASME's standards activities is not limited to IWGs. Our IWG members, however, find the IWG arrangement best facilitates their participation.*

BULLETIN: Do you anticipate that more countries will get involved?

Sanna: *Absolutely. Keep in mind that ASME standards are used in over 100 countries, and there are quite a large number of people around the world with technical talent and experience in applying ASME standards. As we continue our outreach, we expect the number of international participants will continue to increase, within new IWGs as well as on other committees.*

BULLETIN: What is the most significant achievement at this stage of the IWG program?

Crane: *Substantial achievements within a voluntary consensus process can sometimes take several years. While our first IWGs were established in 2009, many are newer, and the integration of IWGs with their standards-writing peer groups is ongoing. In many ways, much of the near-term success we're celebrating is the establishment of IWGs as a platform to support robust activity in the future.*

That said, there have been a number of contributions from IWGs, including proposed standards revisions, interpretations, and "code cases." Each one reflects a technical issue or opportunity that was *identified by the members of the IWG. The proposed interpretations, for example, arose from standards users in the same country – some perhaps from the IWG members themselves – who were experiencing difficulty applying the standard as it is currently published.*

Regardless of whether an issue is universal or unique to another country, the most notable benefit – aside from the additional volunteer-member resources – is that IWGs provide an accessible, local platform for the code user to discuss and begin resolving an issue. That accessibility helps the standards users and IWG members (representing all our standards writers) more easily learn from each other. In fact, sometimes issues are based on a misunderstanding and then settled with a simple discussion. While our committees are always open to addressing issues, IWGs are enhancing their ability to connect with their stakeholders.

BULLETIN**:** How do you see IWGs evolving?

Sanna: *One interesting possibility we're looking into is joint IWG meetings. With multiple, related IWGs in the same country, there is an opportunity to meet in conjunction with each other, providing for shared discussion and exchange of information. This is particularly interesting because many of our standards committees (the "parent committees" of the IWGs) have operated in this manner for many years. For example, the standards committees associated with the ASME Boiler & Pressure Vessel (BPV) Code have, along with most of their subordinate groups, traditionally convened meetings together four times per year, during what we refer to as "BPV Code Week." It would be very interesting to see a number of IWGs in China, all affiliated with the BPV Code, to meet together to create similar synergies.*

Another interesting possibility is that some IWGs may expand beyond a single group. Our IWGs in China – some with nearly 40 members each – are approaching a size where managing the group's discussions at meetings may become a challenge for the chairs. As more individuals express interest in participating in IWGs, we're considering forming smaller groups beneath them – perhaps "task groups" – to accommodate the additional members.

BULLETIN**:** What's involved in your roles as liaisons?

Crane: *Our roles as liaisons correspond to activities of the Nuclear Codes and Standards Department. ASME's volunteer committee members contribute their technical expertise to our standards development activities, but most interactions between ASME Standards and Certification (S&C) and its stakeholders and partners are conducted by staff.*

Nearly all S&C "technical" staff have responsibilities that require travel, but the nature of the global nuclear industry –specifically the use of ASME's nuclear standards in many countries – calls for a number of activities and initiatives outside of the US. Working with the nuclear IWGs is among those activities, and we have been attending many of their meetings to facilitate their integration and success.

BULLETIN**:** Has working with IWGs made an impact on you?

Sanna: *Our volunteer-members are remarkably dedicated and knowledgeable, and working with them – regardless of where they're from – is always a great honor and an opportunity for us to learn. It's no different with our IWG members, who are happy to share perspectives which may be new to us. On a personal level, many of our volunteer-members have become friends and cherished mentors.*

BULLETIN**:** Thank you, Mr. Crane and Mr. Sanna. We wish you continued success in the growth of the IWG program.

Christian Sanna is a Project Engineering Manager at ASME, supporting the development and implementation of ASME's Nuclear Codes & Standards, and is the Nuclear Codes & Standards Asia Liaison. He is currently a member of the BPV III Executive Committee, the BPV III Special Working Group on New Advanced LWR Construction Issues, and the Committee on Board Strategic Initiatives. He was a member and staff Secretary for the ASME BPV Section III Committee on Construction of Nuclear Facility Components for 21 years, and was a member and staff secretary for the ASME Committees on Nuclear Quality Assurance and Nuclear Air and Gas Treatment.

Ryan Crane, P.E., is employed by ASME with 16 years of experience in the areas of codes, standards, conformity assessment and certification programs, and international workshops and conferences. He manages codes and standards activities in a variety of fields, including nuclear power; system energy assessment; reliability, availability, and maintainability of power plants; and verification and validation of computational modeling and simulation. Mr. Crane also serves as the European Liaison for ASME's Nuclear Codes and Standards Department and is active with a variety of international organizations and standards developers .

Certification of Nuclear Pressure Relief Valves

BY JOSEPH F. BALL, P.E., DIRECTOR, PRESSURE RELIEF DEPARTMENT

During recent National Board General Meetings and *American Society of Mechanical Engineers* (ASME) *Boiler and Pressure Vessel Code* meetings, there were a number of presentations on the causes and outcomes of the March 2011 accident at the Fukushima Daiichi Nuclear Power Plant in Japan. One featured a graph of plant system pressure versus time over the first several days of the accident. At certain locations on the graph, the pressure line rises and then becomes flat. The presenter indicated that was when the plant's pressure relief valves (PRVs) were operating. Although there was tremendous damage to the facility and environmental contamination, the almost unthinkable outcome of a nuclear reactor vessel pressure explosion was avoided due to the operation of those PRVs. To me, this points out the importance of PRV equipment and the vital role it plays in nuclear plant safety when other equipment does not function as intended.

This article reviews similarities and differences in the design and certification of nuclear service PRVs and the National Board's role in the certification process.

Since PRVs provide the same function in nuclear and non-nuclear systems, the basic operation of nuclear PRVs is the same as non-nuclear valves. In fact, some manufacturers' designs are certified for nuclear service and other applications, such as boiler or pressure vessel overpressure protection.

Differences are found in the mechanical design of PRVs. Design inputs such as pressure forces, reaction forces, and piping loads are supplied by the system designer and the valve manufacturer must provide a detailed pressure containment design to ensure that allowable stresses are not exceeded. Depending on materials and valve size, additional non-destructive examination (NDE) may be mandated for certain materials, such as castings and forgings, often used for the valve body material. For some nuclear valves, the valve disk position must be verifiable, which leads to the addition of proximity switches (which sense the position of the valve stem during valve operation).

Outlet piping is often taken to a safe point of discharge through long pipe lengths, causing significant amounts of back pressure. In this instance, a balancing piston or bellows will be provided to protect the valve from the effects of back pressure.

Power-operated PRVs may be used for some nuclear applications. The valve is triggered by a calibrated pressure switch that actuates an energy source, such as compressed air, to open the valve. These valves are often used for other functions in the system besides overpressure protection, such as startup circulation, where the valve can be manually actuated from the control room.

Quality assurance during the manufacture of nuclear PRVs is held to the highest levels, as demanded by nuclear standards. A key difference in the quality assurance process is the requirement to have a third party authorized nuclear inspector (ANI) involved in the manufacturing process. The ANI will identify key hold points during manufacturing and witness operations such as pressure testing, inspection of incoming material, weld fit-up, identification of material markings, and performance testing. Work on each PRV is documented on an ASME NV-1 data report. Before the data report can be signed, it must be verified that capacity certification testing was completed and certified by the National Board.

The National Board is the ASME-designated organization responsible for the capacity certification program, which uses rules from the ASME *Boiler and Pressure Vessel Code*, Section III, Subsections NB through NE (Article NX-7000). Although the program is similar to the certification process used for other code sections, there are some differences because of the unique environment within the nuclear codes.

The first difference is that product recertification is not actually mandated. The ASME code was envisioned for the production of components for specific nuclear power plants, and manufacturing of valves as a "catalog" item (where the same design is produced over a long period of time) is not addressed. The National Board, however, requires that each Section III capacity certification be re-registered every six years. Design changes may necessitate additional

Photo courtesy of National Technical Systems.

Farris Engineering's power-operated pressure relief valve under full-flow test at National Technical Systems (NTS) in Huntsville, Alabama.

testing. For designs that are certified for more than one ASME Code Section (Sections III and VIII), the design must pass re-certification testing for each Code Section to remain certified.

Capacity certification for nuclear low-pressure applications (below Sections I and VIII minimum pressure scopes of 15 psig) is mandated, and sub-critical flow equations appropriate for this pressure region are included in the nuclear codes. Vacuum relief valves are also in the nuclear codes (again, not included in Sections I and VIII). Low-pressure relief valves and vacuum relief valves are primarily used on storage tanks. Vacuum relief valves may also be needed for the reactor containment vessel.

The Section III rules do anticipate that some of the valves for nuclear applications can be very large and have high set pressures or high capacities that would exceed ASME/NB-certified flow laboratory capabilities. ASME rules mandate a steady state flow test in which the valve inlet pressure is increased to a specified overpressure and a flow measurement taken. Steady state flow testing is not possible on main steam valves for boiling water reactors with set pressures from 1,150 to 1,315 psig; six-inch inlets; and capacities over one million pounds of steam per hour (from an economic standpoint, a complete power plant would be needed to test one valve). Pressure relief valves for pressurized water reactors can have set pressures as high as 2,600 psig. Again, building a flow test system for this high pressure is not economically feasible.

To address this problem, the code mandates a two-step process where capacity is determined using flow models, and the function of the valve is demonstrated with actual valves at operating conditions.

During flow model testing, the manufacturer proposes details for three different scale models. Three models are used to demonstrate there are no significant changes between the different model sizes, which assures the capacity will be accurate when scaled up to the full-sized valve. The National Board reviews the models to determine whether they are accurate representations of the actual valves to be built. This review looks at various dimensional ratios describing the flow path, including the ratio of the valve lift to the orifice diameter (L/D ratio), which indicates how far the full-size valve should open. Flow testing is done on the flow models to establish the coefficient of discharge, which is then used to calculate the rated capacity of the valve at service conditions.

Flow models are not functional valves since they are used for flow measurement only. The Demonstration of Function (DOF) test is performed to demonstrate the valve will have acceptable performance at the actual size and set pressure needed in the nuclear power plant. The DOF testing must take into account the inlet pressure drop that will be experienced in the actual system, and the built-up back pressure that can be expected. Also included in the test program are the expected fluid and environmental conditions the valve will be subjected to. These valves are often in a high-temperature location, and this environment is reproduced on the test stand by insulating the valve body, or by putting the valve into an enclosure where hot air is circulated around it.

The valve is tested to demonstrate set pressure and blowdown, and the disk lift is measured. The test equipment needed for the valve to reach full lift consists of a large pressure vessel where the valve is installed, which is supplied by another pressure vessel filled to a higher pressure than the test pressure. Control valves between the two pressure vessels are quickly opened, which increases the test pressure until the valve "pops" open. The inlet pressure continues to increase until the code-specified overpressure is achieved, and then the PRV is allowed to reclose. All test parameters are measured with computer-based data acquisition systems and the data obtained is analyzed.

This test is repeated several times to show consistent performance. Variations in back pressure may be needed to account for different anticipated conditions in service. This testing is witnessed by a National Board representative who determines if performance characteristics have been met. The valves must meet specified performance tolerances and demonstrate the disk lift as indicated in the flow models.

The test program may also include seismic qualification, life cycle performance (repeated set pressure tests), and seat leakage measurements. Power-operated valves may be tested for other functions, such as low-pressure opening performance and response speed. Successful completion of the tests allows inclusion of the design in the National Board publication, *Pressure Relief Device Certifications* (NB-18), showing that certification has been obtained for the particular test fluid, pressure range, and application the valve type was designed for.

Since the capacity and pressure needed for these tests is quite high and the measurement capability very specialized, there are only a few locations throughout the world that can perform this work. I have witnessed a number of these tests at National Technical Systems (NTS) in Huntsville, Alabama, (formerly Wyle Laboratories), which tests using steam and hot water at the desired conditions. Typically during a test project, test personnel, valve manufacturer's engineers and technicians, the National Board test witness, and perhaps a utility representative verify test data and examine test graphs. However, for one or two of the test cycles, the witnesses will go outside of the test center control room to truly experience the test. Although a considerable safe distance is maintained and double hearing protection used, the valve test releases an extremely loud and impressive noise as well as long plumes of steam, both of which demonstrate the immense amount of energy released during the overpressure event.

Witnessing this performance test "live" also reinforces the vital role PRVs play in ensuring safety in a nuclear power system. The National Board recognizes the significant investment a valve manufacturer must make in designing, fabricating, and performing this rigorous testing program. Third-party inspections and the role of the National Board in this verification and testing process contribute to the assurance that nuclear power continues to safely provide energy for countries throughout the world. We can take great pride when safety devices continually function as required to maintain the safety of nuclear facilities over many decades of operation. ✚

Test Number 40,000 Completed at National Board Testing Lab

The National Board Testing Laboratory completed its 40,000th test on May 27, 2015. The test was performed on an air service valve manufactured by HEROSE GmbH Armaturen of Bad Oldesloe, Germany. The valve successfully met all requirements.

"This achievement represents many hours of work on behalf of staff in our Pressure Relief Department over two decades," commends Executive Director David Douin. "And consider this," he continues, "Each of those 40,000 tests actually may represent hundreds or thousands of certified pressure relief devices operating on equipment around the world. We at the National Board are very proud of this accomplishment."

The lab's Pingue Drive location in Columbus, Ohio, opened in 1991. Since then, each test performed at this location has been tracked sequentially. The facility is the world's only independent ASME-certified flow laboratory and is equipped with three test systems that use steam, nitrogen, and water as the test media. Each year the lab works with manufacturers, assemblers, and repair organizations from around the globe to test the performance and relieving capacity of pressure relief devices. ♣

ABOVE From left to right: National Board lab engineers Austin Peck and Bob Viers with HEROSE representatives Marc Zaubitzer and Martin Boyungs.

RIGHT Test number 40,000: HEROSE air service valve.

Nuclear Inspectors: A Few Key Distinctions

Thinking of a career in nuclear inspection? Two National Board associates, senior staff engineer Bob Ferrell and consultant Walter Beach, sat down with the BULLETIN and discussed some distinctions of nuclear inspection.

Photo courtesy of Nuclear Regulatory Commission.

NRC Senior Resident Inspector Silas Kennedy performs a routine inspection at the Calvert Cliffs Nuclear Power Plant in Lusby, Md. (May 2012)

Nuclear endorsements follow a defined progression. Those seeking a nuclear endorsement must start with the New Construction Authorized Inspector **(A)** endorsement and fulfill one year of experience in that role before pursuing the Authorized Nuclear Inspector **(N)** endorsement. The **N** endorsement then becomes the foundation for all other nuclear endorsements.

The **A** endorsement mandates some basic foundations in new construction, such as metallurgy, welding qualifications (ASME IX), non-destructive examination (NDE [ASME V]), inspection requirements, and duties of an inspector to ASME non-nuclear new construction codes (the ASME *Boiler and Pressure Vessel Code* [B&PVC]).

In qualifying for the **N** endorsement, nuclear inspectors work specifically with ASME Section III, *Rules for Construction of Nuclear Facility Components,* and use the reference codes of welding (Section IX) and NDE (Section V) as applicable to specified nuclear requirements. The student uses some of the fundamentals learned in the **A** endorsement as the basis for duties and responsibilities and applies them to the requirements specified in the nuclear construction codes.

Once inspectors meet the **N** endorsement course requirements, they can pursue the Authorized Nuclear Inservice Inspector **(I)** endorsement, which uses ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components.* This code not only addresses inservice inspection but also covers repair/replacement activities in nuclear power plants.

Next in the progression is the Authorized Nuclear Inspector (Concrete) **(C)** endorsement. This focus is primarily for the protection/containment of the plant's reactor, which is designed to withstand the crash of a jet airliner. Concrete materials are highly controlled through documentation, inspection, and testing, and there are stringent requirements for reinforcement and placement, which is why there is a specialized nuclear inspector **C** endorsement.

A specific minimum of diversified work experience is required within each **N**, **I**, and **C** endorsement in order to qualify, and there are supervisory levels for each endorsement that can be pursued: Authorized Nuclear Inspector Supervisor (**NS**), Authorized Nuclear Inservice Inspector Supervisor (**NSi**), and Authorized Nuclear Inspector Supervisor (Concrete) (**NSc**).

Domestically, the United States Nuclear Regulatory Commission (NRC) enforces regulations that are mandated by the federal government. In nuclear inspection and operation, the federal government holds the owner accountable, as well as any others involved, in regulatory violations. Nuclear inspectors who violate rules or regulations can potentially face criminal prosecution, conviction, and possible jail time. Non-nuclear inspections are mandated by US states and Canadian provincial jurisdictions, which hold the owner liable for violations. The most severe consequence for non-nuclear inspectors is to have their commissions revoked as a result of neglect of duties.

Both nuclear and non-nuclear inspections require detailed accounts and observations, but the level of documentation increases in nuclear due to the fact that nuclear quality systems are more complex and involved. Bob Ferrell recalls an insider joke: "When the paperwork weighs as much as the item, you have enough paperwork." Part of this is due to the stronger emphasis on material records, such as material verification, traceability, and pedigree; as well as tracking what tests occurred; maintaining lab certifications/calibration records; and performing longer, more comprehensive audits (some can take up to two weeks). All of these activities result in a greater amount of paperwork that must be maintained.

Another point: some nuclear shops focus on machined parts (no welding at all) and some shops weld the parts together. Nuclear codes include castings, forgings, and other materials, which means that even mass-produced manufactured parts must meet the required material certification. In fact, every piece of equipment that contains or supports systems within the reactor containment – even radioactive waste storage – is required to meet design, construction, inspection, examination, testing, and certification requirements of the ASME nuclear codes.

The refueling outage is the busiest time for authorized nuclear inservice inspectors (ANIIs). When a nuclear plant is operating, very little testing and examinations are performed. That changes when the plant goes into a refueling outage, which happens about every 18 months. During a refueling outage, a critical path plan is put together and all of the inspections fall in line to meet code requirements. The inspector is very busy during this time, and authorized inspection agencies (AIAs) might send in more than one inspector.

Authorized nuclear inspectors (ANIs) have the right to identify and report problems without repercussion from their employers. The federal document, NRC: 10 CFR Part 21 – *Reporting of Defects and Noncompliance,* states that if inspectors see problems in a nuclear plant that aren't being addressed, they can call the NRC and report it with no repercussion. Companies domestically are required to teach and post this procedure. Non-nuclear inspectors do not have this provision for reporting problems.

Authorized nuclear inservice inspectors (ANIIs) and authorized nuclear inservice inspector supervisors (ANIISs) must be aware of radiation hazards. Inspector safety is a top priority across nuclear and non-nuclear inspection duties, but the inservice nuclear inspector must factor in radiation safety and become trained in a health physics course, which teaches radiation safety, exposure limits, the different types of radiation, protective clothing requirements, and more. An overview of health physics is taught in the **I** endorsement course, but employers provide plant-specific, practical training for their inspectors.

Authorized nuclear inspectors (ANIs) are audited more often by their supervisors. Non-nuclear **A** inspector audits are mandated once a year, but within nuclear inspection, inspectors at a minimum are audited twice a year. In some instances, a nuclear inspector could be audited more than that depending on the inspector's criteria. For instance, if an inspector works in four different plants, the supervisor must perform an audit for each location.

Nuclear inspectors monitor a certificate holder to a more complex program, which in effect takes more time. Both nuclear and non-nuclear inspectors are required to monitor the certificate holder's quality assurance program; however, due to the complexity of a nuclear quality system, monitoring requires more detailed understanding of quality systems. It also takes more time to complete, and, as already mentioned, more paperwork is involved.

For more information about the National Board nuclear training courses, click the "Training" tab at www.nationalboard.org. ♣

The Revised National Board **NR** Accreditation Program

For Repair and Replacement of Nuclear Components

BY CHUCK WITHERS, ASSISTANT EXECUTIVE DIRECTOR – TECHNICAL

The 2015 Edition of the National Board Inspection Code *brings with it a completely revised* **NR** *program. This article explains why a complete revision of the* ***NR*** *program was needed, how organizations obtain an* **NR** Certificate of Authorization, *and how they can benefit from using this revised National Board accredited program for repair or replacement of nuclear components.*

The National Board **NR** *Certificate of Authorization* was effectively implemented January 1, 1979, for organizations wishing to be accredited by The National Board of Boiler and Pressure Vessel Inspectors (National Board) to repair or replace nuclear components in accordance with a written quality assurance program that covers a specific requested scope of activities. This program includes the quality manual with supporting procedures that address all the controls (who, what, when, where, why, and how) needed to ensure the understanding and application of codes and standards. This program ensures regulatory requirements are met, and, until the 2015 Edition of the NBIC, was patterned after the *ASME Boiler and Pressure Vessel Code* (ASME Code), Section XI.

Why a Complete Revision?

Application for the **NR** *Certificate of Authorization* is available to any qualified organization including owners of nuclear facilities, manufacturers, and nuclear repair organizations. Based on changing needs for constructing nuclear plants using pre-existing stamped components, and clarification of regulatory authority requirements, the National Board, along with the NBIC Committee, elected to completely revise and rewrite the **NR** Accreditation Program.

A driving force for this change was based on a determination made by the United States Nuclear Regulatory Authority for repairing or modifying existing nuclear components. In the past, once an item was stamped in accordance with ASME Code Section III requirements, it was permissible to repair the item to ASME Code Section XI or to the owner's repair program, regardless of location or status of operation. Today, only after fuel loading can ASME Code Section XI or the owner's repair program be utilized for repair or replacement activities. With the onset of new construction, many previously constructed ASME components that were stamped per Section III but not installed, or previously installed in a nuclear facility where construction was halted, are now being used. Over the years, many of these components need repair or modifications. Since ASME Code Section III concerns new construction only, there are no requirements for repairs to components once stamped. Therefore, since ASME Code Section III or Section XI cannot be used to repair or modify existing nuclear components, the **NR** program is an ideal solution.

What is Needed for NR Accreditation?

This revised **NR** accreditation program now provides the flexibility organizations need to comply with codes, standards, and regulations worldwide while maintaining the quality and safety of repaired and replaced nuclear components. Now, applicants for the **NR** *Certificate of Authorization* must address all controls within their quality assurance program based on the category of activity and the scope of work to be performed (organization's capabilities) for which certification is requested. This written quality assurance program is based on at least one of the following three categories depending on responsibilities (an owner or an organization other than the owner).

Category 1 allows for repair or replacement activities to components or systems that have been certified and stamped to ASME Code Section III, irrespective of physical location and installation status prior to fuel loading.

Category 2 allows for repair or replacement activities to items or systems under the scope of ASME Code Section XI, irrespective of physical location.

Category 3 allows for repairs or replacement activities for items constructed according to codes or standards other than the ASME Code irrespective of physical location, installation status, and fuel loading.

An organization's written quality assurance manual for Category 1 will meet the requirements specified in ASME Code Section III, specifically NCA-4000, as well as the regulatory requirements of 10 CFR Part 50 Appendix B.

Category 2, depending on the owner's requirements, may be written to meet the requirements of ASME Code Section XI, IWA-4142, NQA-1 Part 1, or 10 CFR Part 50 Appendix B, and supplemented as needed with the owner's quality assurance program.

The quality assurance program for Category 3 may be written to meet NQA-1 or specify the standard to which certification is desired. These requirements are stipulated in the table below:

TABLE 1.8.2

"NR" QUALITY ASSURANCE PROGRAM (QAP) REQUIREMENTS

Category of Activity	Owner	Organizations other than Owner
Category 1	10 CFR Part 50 Appendix B1, 2, and ASME Section III NCA-4000	10 CFR Part 50 Appendix B1, 2, and ASME Section III NCA-4000
Category 2	10 CFR Part 50, Appendix B1, 2, or NQA-1, Part 1 and ASME Section XI, IWA-4142	10 CFR Part 50, Appendix B1, 2, supplemented as needed with Owner's QA program; or ASME NQA-1, Part 1; or ASME Section III, NCA-4000
Category 3	ASME NQA-1, or specify the standard to which certification is desired	ASME NQA-1,or specify the standard to which certification is desired
Note 1: Code of Federal Regulations (CFR) – rules and regulations published by the executive departments and agencies of the federal government of the United States.		
Note 2: 10 CFR 50 Appendix B – Title 10 of the Code of Federal Regulations Part 50 Appendix B describes the quality assurance criteria for nuclear plants and fuel reprocessing plants.		

Some prerequisites for any organization to obtain a National Board **NR** *Certificate of Authorization* include the following:

- Have and maintain an inspection agreement with an authorized nuclear inspection agency.
- Have a written quality assurance program addressing all the controls for the intended category and scope of activities.
- Have a current Edition of the NBIC.
- Have available the codes and standards appropriate for the scope of work to be performed.

Additional administrative requirements can be found on the National Board website, under "Stamps and Marks," then "**NR** Stamp," procedure NB-417.

Before an **NR** *Certificate of Authorization* is issued, the applicant must have its quality assurance program and the implementation of the program reviewed and found acceptable by a survey team composed of qualified representatives from the National Board, the jurisdiction, and the authorized nuclear inspection agency.

An acceptable written quality program and its implementation will ensure quality repair and replacement activities are performed and maintained on nuclear components, items, parts, and systems. These rules are the basis for evaluating each quality assurance program prior to the issuance or renewal of a National Board **NR** *Certificate of Authorization*. This accreditation process is identical to the ASME survey performed to issue a *Certificate of Authorization* for a nuclear organization to use ASME's Certification Mark.

IMPORTANT: Each organization must now clearly describe the category of interest, scope of capabilities, and controls needed for each category.

How Organizations Can Benefit

This complete enhancement to the original **NR** Accreditation Program provides numerous benefits for owners, regulatory authorities, and other organizations involved in nuclear repair and replacement activities. Among the main benefits are:

- The **NR** program is accredited and developed under a consensus process, is easily recognized, and can be adopted readily by organizations throughout the world.
- This accredited program may be used to ensure quality regardless of codes or standards used for construction, location of installation, or certification.
- The National Board and the *National Board Inspection Code* are recognized worldwide for promoting principles of safety and uniformity of pressure equipment.
- The **NR** program is reviewed and evaluated every three years to ensure the quality program is updated, understood, and implemented as described within the written quality manual.
- The program provides a means to document and certify repairs and replacement activities on forms registered with the National Board.
- National Board accredited quality repair programs are easily recognized by applying the conformity assessment symbol stamp.
- Quality and conformity are verified and monitored through an accredited authorized nuclear inspection agency using valid and qualified National Board commissioned nuclear inspectors.
- The program provides for availability, retention, distribution, and retrieval of documentation.
- The program provides a means of investigating organizations and inspectors and taking disciplinary actions when codes or standards are not met.
- Finally, licensee costs can be minimized by understanding that:
  1. Code-required audits of repair organizations are not needed since periodic review of certificate holders is performed consistently every three years by the National Board Survey Team.
  2. Certificate holders assume survey costs.
  3. The licensee may obtain an **NR** *Certificate of Authorization* and have repair organizations work under its quality program.
  4. An accredited program easily identifies qualified repair organizations when a *Certificate of Authorization* is issued and maintained.
  5. The accredited program minimizes time needed to survey and audit to qualify repair organizations.

It is the hope of the National Board that worldwide recognition of the revised National Board **NR** Accreditation Program will serve to **unify** nuclear repair and replacement activities.

When organizations follow this proven, recognized, and accredited quality assurance program, the nuclear industry worldwide can be assured that acceptable codes and standards are followed; jurisdictional and regulatory requirements are understood and followed; and in-service inspection, testing, and repair methods are applied satisfactorily. We can all benefit from knowing that properly repaired or replaced nuclear pressure-retaining items will perform as originally designed and will continue to operate safely.

The National Board of Boiler and Pressure Vessel Inspectors is accredited by the American National Standards Institute (ANSI) as a Standards Developing Organization (SDO). Since 1945, the National Board has developed one internationally recognized post-construction standard known as the National Board Inspection Code *(NBIC). Part 3,* Repairs and Alterations, *contains administrative and technical requirements of the* **NR** Certificate of Authorization. *The National Board* ***NR*** *Accreditation Program is one of three accreditation programs administered by the National Board.* ✚

Transparency

BY JAMES R. CHILES

Even as attention and excitement focus on the new generation of power reactors, it's good to look back before leaping forward. One lesson comes from how then-Captain Hyman G. Rickover got the Nuclear Navy off the ground and into the water.

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In late summer 1954, contractors were trying to meet the captain's deadline of starting the sea trials of *Nautilus*, the world's first nuclear-powered vessel, before New Year's Day. But on September 16, a line in the non-radioactive steam loop blew apart and investigation showed that the piping involved, and maybe much more piping all around the crowded reactor room, had been fabricated from the wrong alloy. Somehow, a contractor made a serious error, and double-checking by Navy men had not caught the problem.

What to do? Some managers might have hushed up the problem as an embarrassment, done a quick cost-benefit study in hopes of limiting the rework needed, and concocted a plausible excuse for missing the deadline. Instead, Captain Rickover had all the suspect piping ripped out and replaced, even though it set the program back three months. What's even more remarkable is that Rickover made sure everyone in the program knew what had happened. Six decades later, that piping problem and the Nuclear Navy's response still stands as a beacon of world-class quality.

And a beacon of transparency. His reasoning: carelessness with reactors and peripheral equipment not only risked the lives of sailors and officers, but could destroy public trust that was essential if the new fleet were to be welcomed into harbor cities around the world. Lose trust; lose the program. To Rickover, it was that simple. No half-measures would suffice.

Rickover's view of the big picture is as sound as ever: nuclear power is radically different from any other forms of engineering. It works on longer timescales, requires bigger investments, and offers greater rewards; but carelessness can pose massive risks. It calls engineers to new heights of performance. Rickover knew from the get-go that harnessing the atom would take a new and dead-serious mindset: "The discipline of technology," he called it.

Brutal honesty about performance is vital to that discipline. Take this portion of the 2011 preamble to the voluntary Code of Conduct signed by nine major nuclear-technology exporters, the product of long discussions led by the Carnegie Endowment: "... to advance public confidence by upholding high standards of transparency, integrity, ethical behavior, and social responsibility, and to promote continuous improvement toward the implementation of global best practices."

Transparency is often a shorthand term for disclosure to citizens and politicians. But here I want to focus on transparency of a more limited kind: what goes on *inside* the reactor community – the free sharing of lessons learned and best practices between reactor operators, designers, and regulators – what we might call "internal transparency," which can be very valuable even when chunks of that information never reach the outside world.

An example is how the Institute of Nuclear Power Operations (INPO) keeps its operator reviews confidential. There's been a similar effort to keep international operators informed through the World Association of Nuclear Operators. Brand-specific operator groups add more value to the mix. After concerns about Russian water–water energetic reactors (WWER) and graphite-moderated, water-cooled (RBMK) reactors, plant operators and Russian designers teamed up to produce a valuable series of type-specific safety advisories in the 1990s called *Issue Books*.

The rapid rise of internal transparency started just weeks after the partial meltdown at Three Mile Island Unit 2 (TMI-2), and is an important reason why the worldwide fleet of 400-plus reactors (mostly Generation II models, using light water as a moderator) in large part overcame a major mid-life crisis. According to a widely read 1985 article in *Forbes,* titled "Nuclear Follies," excessive facility cost was going to strangle the industry, driven by a multitude of causes from red tape to poor operating practices to bad financial management. "For the U.S.," wrote James Cook in *Forbes*, "nuclear power is dead – dead in the near term as a hedge against rising oil prices and dead in the long run as a source of future energy. Nobody really disputes that."

Now we know that a great many nuclear power plants of Generation II vintage turned out so profitable that owners want to extend their lives as long as possible.

How did that happen? Prompt, thorough exchange of lessons learned was one reason, and conditions were favorable. Most Generation II plants were of the same general type, using light water as a moderator, and all were under some public skepticism for a time, so operators were in the same lifeboat and rowing in the same direction. Through most of this post-Chernobyl, post-TMI-2 period, competitive rivalries receded and cooperation advanced. What's more, the exchange of information went beyond accident-lessons-learned to best practices: "Here's what we did to improve quality and bring down costs, too."

Yes, there were gaps and pauses in the flow of information, and some near misses, but improved transparency certainly played a part in the remarkable comeback. That comeback is verified by the reactors' higher production, lower random-outage rates, and operational timelines that could hit 80 years.

Accidents at Three Mile Island Unit 2 and Chernobyl brought about the formation of the INPO in the US, and the World Association of Nuclear Operators (WANO) internationally. INPO and WANO, along with regulators, still do their job of sharing lessons learned about problems, but how much information sharing will we see when it comes to a new wave of reactor alternatives, produced by a half-dozen or more companies around the world?

The aftermath of the Chernobyl meltdown.

Companies will have to hustle, according to remarks Russian president Vladimir Putin made while meeting with French business leaders in February 2013: "The Rosatom state corporation, together with its French partners, works on developing a fourth-generation fast neutron reactor. I feel that such joint efforts will give Russian and French nuclear experts a tangible competitive advantage on the global market, where the competition is very tough and is constantly increasing."

About that new generation. We're told that the coming boom in new-style Generation IV reactors, which could greatly expand the world's population of

Three Mile Island.

power reactors by 2050, won't increase risk to the public. We're told panels of international experts have screened proposals and identified a short list that will meet strict criteria for safety, efficiency, and affordability. One criterion is a passive system for safe shutdown that doesn't rely on immediate, external supplies of water and power for emergency cooling. These new reactors will be fundamentally safer than the headline-making ones in Pennsylvania, Ukraine, and Japan – and more affordable to build and operate. They'll be standardized and simplified.

Those are all fine goals, but here's my question: Will the energetic competition between reactor builders lead to a trade-secret "silo" attitude that could hinder the flow of information? My question is not so much about the early years, when designers reasonably expect to profit from their advances in materials and equipment, and before operational experience builds up. Later on, post-construction, will data from the field be slapped with a proprietary label, not so much for profit but a misplaced sense of self-protection? If it is, we could see another midlife crisis of the kind that the Generation II light-water reactors faced.

Hiding problems behind claims of proprietary information has happened before, as the following examples show:

Following the partial meltdown at Three Mile Island Unit 2 in 1979, no public disclosure was made about why the relief valve atop the pressurizer stuck in the "open" position. Court records were sealed as part of a settlement agreement.

After a serious fuel-cleaning incident at the Paks nuclear power plant in Hungary in 2003 (which required several years to clean up), the operator resisted releasing some technical information to state regulators, claiming the technical details were proprietary.

Noting in a *Nuclear Plant Journal* article in 2010 that WANO gets event reports from 400 nuclear plants worldwide, and that publicizing the lessons learned is fundamental to the organization's purpose, WANO President Laurent Stricker went on to express this concern: "There are a few nuclear plants who do not share much information. WANO works directly with the chief executives of these utilities to convince them that the lack of transparency and openness to share events information is a weakness for the whole nuclear industry. WANO is the best tool for the CEOs to ensure the global nuclear safety of all the plants. The weakest link in the nuclear power industry can affect the entire nuclear power industry worldwide."

In 2009, Exelon Nuclear Corp. declined to make a root-cause study public about tritium releases from the Oyster Creek Nuclear Generating Station in New Jersey, citing the need to protect proprietary information. After discovering that a summary in an op-ed piece did not address public concerns, Exelon released the documents.

One promising Generation IV reactor that has made the short list of promising concepts is the Very High Temperature Reactor (VHTR), which in addition to serving power generation could supply process heat to replace the huge quantities of oil, coal, and gas now consumed by heavy industry. And VHTRs could provide fossil-free supplies of hydrogen for clean transportation.

Members of the public may wonder why we don't have big VHTRs running now, given that graphite-moderated, gas-cooled reactors have been tried at smaller scales. One reason that interests me is the need for major leaps in materials science, which could be the subject of much wrangling about trade secrets. We'll need highly advanced materials for the reactor core, perhaps ceramics, that will hold up under long exposure to very harsh conditions of temperature and neutron bombardment. Just an hour browsing trade journals will indicate how much work needs to be done to come up with materials that can reliably tolerate high temperatures and powerful neutron bombardment. It's not optional: it's what the VHTR will need to cut costs and achieve high fuel burn-up rates.

Here's a question. Let's say several competing VHTR models go into service, from China, Japan, the US, and elsewhere. What if one builder comes across unexpected behavior that is relevant to the safety of other designs – something that's not yet a reportable incident, but more like a trend? How will that information be exchanged, and will there be a long delay in transmission due to trade-secret concerns – a delay so long that poor design choices go forward, unchallenged?

Early in Generation IV discussions, Shannon Burke of the American Society of Mechanical Engineers (ASME) pointed out that a combination of intellectual property factors and a shortage of materials experts was going to make it difficult to agree on a consensus standard for advanced materials. As Norman Hilberry, then-director of Argonne National Laboratory, famously said in the mid-1950s: "We physicists can dream up and work out all the details of power reactors based on dozens of combinations of the essentials, but it's only a paper reactor until the metallurgist tells us whether it can be built and from what. Then only, one can figure whether there is any hope that they can produce power."

Going to the larger issue, transparency beyond trade-secret concerns, what are the best conditions for fast and thorough information transfer? We could list factors such as standardization of plants, good training, and expert interpretation of incident reports. All those are important. But they won't be enough without the right attitude: a fundamental willingness to reach across business-as-usual barriers in the common interest of extraordinary safety.

My concern about this potential downside of proprietary problems among newly competitive markets for nuclear reactors may never come to pass. But my crystal ball predicts that the public and press will be alert for any signs of carelessness or cover-ups, whether the generations are dubbed II, III-Plus, or IV. Vendors, operators, and regulators who think they can hide embarrassing facts behind trade-secret blankets must remember that just one major accident, anywhere in the world, can yank those screens away.

And citizens have a right to speak their minds. Reactors aren't like aging airplanes, ships, and trucks, which tend to fade from view, driven by rising upkeep and fuel costs. Baseload nuclear reactors aren't going anywhere until decommissioned and demolished. If regulators get lax, marginal units may be a problem long after their best years have passed. So while I admit that we reporters and citizens don't need to know everything about a reactor, and shouldn't get in the way of professional information transfer among industry players (the "internal transparency" above), we do deserve best-practices safety and defense-in-depth designs.

Yes, Generation IV investors will have billions of dollars at stake; yes, workers inside the fence will have their lives on the line if something goes wrong. But lots of other people have a direct stake too, like many thousands of people who live downwind. As Hyman Rickover told anyone who'd listen, reactors are different from other machines and demand nothing less than Rickover-level safety, the discipline of technology. ✤

2014 Report of Violation Findings

The National Board Annual Violation Tracking Report identifies specific violations (per device type) commonly found on five types of pressure equipment during jurisdiction-required inspections. The following data reflects the reporting period of 1/1/2014 – 12/31/2014 as reported by participating member jurisdictions.

The Violation Tracking Report indicates problem areas and trends related to boiler and pressure vessel operation, installation, maintenance, and repair. The data also identifies problems before adverse conditions occur. This report serves as an important source of documentation for jurisdictional officials, providing statistical data to support the continued funding of inspection programs.

Overall Totals for Each Type of Pressure Equipment

Type of Pressure Equipment	Total Number of Inspections	Total Number of Violations	Percent of Violations
High-Pressure/High-Temperature Boilers (S)(M)(E)	72,279	5,129	7.10%
Low-Pressure Steam Boilers (H)	49,546	8,570	17.30%
Hot Water Heating/Supply Boilers (H)	266,992	35,743	13.39%
Pressure Vessels (U)(UM)	223,081	7,273	3.26%
Potable Water Heaters (HLW)	52,089	5,483	10.53%
Totals	**663,987**	**62,198**	9.37%

Number of Jurisdictional Reports: 117

High-Pressure/High-Temperature Boilers (S)(M)(E)

Device Type	Number of Violations	*V/I	**V/V
1) Safety Relief Devices	**763**	**1.06%**	**14.88%**
2) Low-Water Cutoffs/Flow Sensing Devices	**256**	**0,35%**	**4.99%**
3) Pressure Controls	**158**	**0.22%**	**3.08%**
4) Temperature Controls – Operator or High Limit	**41**	**0.06%**	**0.80%**
5) Burner Management	**626**	**0.87%**	**12.21%**
6) Level Indicators – Gage Glasses, Bulls Eyes, and Fiber Opticals	**266**	**0.37%**	**5.19%**
7) Pressure/Temperature Indicators	**106**	**0.15%**	**2.07%**
8) Pressure-Retaining Items (PRI) / Boiler-Piping, Pumps, Systems Valves, Expansion Tanks	**2,913**	**4.03%**	**56.79%**

Summary:

- Number of Jurisdiction Reports: 117
- Total Number of Inspections: 72,279
- Total Number of Violations: 5,129
- Percent Violations: 7.0%

***V/I** Total number of Violations for Category divided by Total Inspections for this device type

****V/V** Total number of Violations for Category divided by Total Violations for this device type

Low-Pressure Steam Boilers (H)

Device Type	Number of Violations	*V/I	**V/V
1) Safety Relief Devices	1,229	2.48%	14.34%
2) Low-Water Cutoffs/Flow Sensing Devices	653	1.32%	7.62%
3) Pressure Controls	616	1.24%	7.19%
4) Temperature Controls – Operator or High Limit	134	0.27%	1.56%
5) Burner Management	1,187	2.40%	13.85%
6) Level Indicators – Gage Glasses, Bulls Eyes, and Fiber Opticals	630	1.27%	7.35%
7) Pressure/Temperature Indicators	279	0.56%	3.26%
8) Pressure-Retaining Items (PRI) / Boiler-Piping, Pumps, Systems Valves, Expansion Tanks	3,842	7.75%	44.83%

Summary:
- Number of Jurisdiction Reports: 117
- Total Number of Inspections: 49.546
- Total Number of Violations: 8,570
- Percent Violations: 17%

***V/I** Total number of Violations for Category divided by Total Inspections for this device type

****V/V** Total number of Violations for Category divided by Total Violations for this device type

Hot Water Heating/Supply Boilers (H)

Device Type	Number of Violations	*V/I	**V/V
1) Safety Relief Devices	7,447	2.79%	20.83%
2) Low-Water Cutoffs/Flow Sensing Devices	2,147	0.80%	6.01%
3) Pressure Controls	169	0.06%	0.47%
4) Temperature Controls – Operator or High Limit	2,742	1.03%	7.67%
5) Burner Management	5,692	2.13%	15.92%
6) Level Indicators – Gage Glasses, Bulls Eyes, and Fiber Opticals	730	0.27%	2.04%
7) Pressure/Temperature Indicators	1,201	0.45%	3.36%
8) Pressure-Retaining Items (PRI) / Boiler-Piping, Pumps, Systems Valves, Expansion Tanks	15,615	5.85%	43.69%

Summary:
- Number of Jurisdiction Reports: 117
- Total Number of Inspections: 266,992
- Total Number of Violations: 35,743
- Percent Violations: 13%

***V/I** Total number of Violations for Category divided by Total Inspections for this device type

****V/V** Total number of Violations for Category divided by Total Violations for this device type

Pressure Vessels (U) (UM)

Device Type	Number of Violations	*V/I	**V/V
1) Safety Relief Devices	3,236	1.45%	44.49%
2) Low-Water Cutoffs/Flow Sensing Devices	2	0.00%	0.03%
3) Pressure Controls	24	0.01%	0.33%
4) Temperature Controls – Operator or High Limit	3	0.00%	0.04%
5) Burner Management	26	0.01%	0.36%
6) Level Indicators – Gage Glasses, Bulls Eyes, and Fiber Opticals	9	0.00%	0.12%
7) Pressure/Temperature Indicators	199	0.09%	2.74%
8) Pressure-Retaining Items (PRI) / Boiler-Piping, Pumps, Systems Valves, Expansion Tanks	3,774	1.69%	51.89%

Summary:
- Number of Jurisdiction Reports: 117
- Total Number of Inspections: 223,081
- Total Number of Violations: 7,273
- Percent Violations: 3%

***V/I** Total number of Violations for Category divided by Total Inspections for this device type

****V/V** Total number of Violations for Category divided by Total Violations for this device type

Potable Water Heaters (HLW)

Device Type	Number of Violations	*V/I	**V/V
1) Safety Relief Devices	1,411	2.71%	25.73%
2) Low-Water Cutoffs/Flow Sensing Devices	109	0.21%	1.99%
3) Pressure Controls	4	0.01%	0.07%
4) Temperature Controls – Operator or High Limit	72	0.14%	1.31%
5) Burner Management	1,082	2.08%	19.73%
6) Level Indicators – Gage Glasses, Bulls Eyes, and Fiber Opticals	2	0.00%	0.04%
7) Pressure/Temperature Indicators	722	1.39%	13.1%
8) Pressure-Retaining Items (PRI) / Boiler-Piping, Pumps, Systems Valves, Expansion Tanks	2,081	4.00%	37.95%

Summary:
- Number of Jurisdiction Reports: 117
- Total Number of Inspections: 52,089
- Total Number of Violations: 5.483
- Percent Violations: 11%

***V/I** Total number of Violations for Category divided by Total Inspections for this device type

****V/V** Total number of Violations for Category divided by Total Violations for this device type

The 84th General Meeting

COLORADO SPRINGS, COLORADO – 2015

Highlights

Photos by Brandon Sofsky

Surrounded by The Broadmoor's impeccable accommodations and Colorado's bigger-than-life landscape, attendees and guests of the 84th National Board/ASME General Meeting in Colorado Springs enjoyed a busy week of both business and pleasure that included technical presentations, networking, entertainment, ASME code and National Board member meetings, and a dash of sightseeing.

On Monday morning, Denver-based string quartet Spinphony and their unique blend of pop and classical music set the tone for the Opening Session before Hollywood icon James Caan took the stage and shared off-the-cuff anecdotes from his career. Monday afternoon, six industry professionals presented at the General Session: Bernard Hrubala (ASME), Patrick Jennings (Hartford Steam Boiler), Carey M. Bilyeu (Zurich), Nathaniel Gee (US Bureau of Reclamation), James F. Reilly (NASA retired astronaut), and A. Thomas Roberts (MPR Associates Inc.). During the General Session, guests of attendees visited the US Olympic Training Center for a special athlete-led tour of the facilities.

National Board members attended a general discussion session and members' meeting on Tuesday. Guests enjoyed a bus tour through Garden of the Gods before being transported to the Royal Gorge Railway, where they traveled on a specially chartered train that traversed the gorge along the scenic Arkansas River.

On Wednesday, guests and attendees enjoyed a casual day of activities, music, and food at Spruce Mountain Ranch before heading back to The Broadmoor for an evening of special recognition, fine dining, and entertainment at the Wednesday Evening Banquet. June Ling was honored with the National Board Safety Medal and comic impressionist Jeff Tracta entertained with his lively mixed-media show.

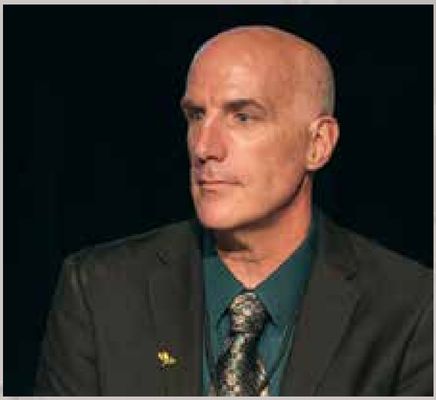
Ling Recipient of 2015 Safety Medal

June Ling, then-ASME associate executive director for standards and certification and longtime friend of the National Board, was awarded the 2015 National Board Safety Medal. The award was presented by National Board Executive Director David Douin at the Wednesday Evening Banquet. ASME colleagues Tom Pastor, Laura Hitchcock, and Madiha Kotb also shared personal and professional remarks regarding Ling's outstanding career at ASME, which spans over 40 years.

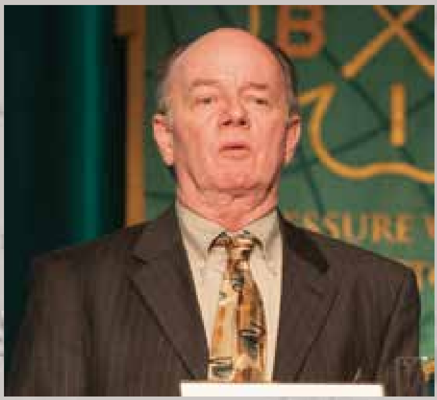
David Douin & June Ling, Right From Top: Madiha Kotb, Laura Hitchcock, Tom Pastor

Board of Trustees Election Results

National Board members voted to fill two expired seats on the Board of Trustees at the Members' Meeting on Tuesday, April 28. Florida Chief Michael Burns was re-elected second vice chairman, and Mississippi Chief Kenneth Watson was re-elected member at large.



Michael Burns

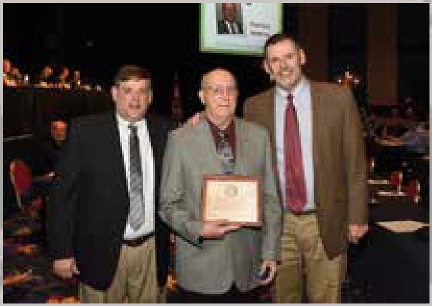


Kenneth Watson

Honorary Members Acknowledged

Donald Jenkins, former member representing Kansas; Keith Rudolph, former member representing Hawaii; and Howard Pfaff, former member representing South Dakota, were bestowed honorary membership status at the General Meeting in Colorado Springs. Mr. Pfaff's posthumous award was presented to his wife and daughter. ✱

John Burpee, Keith Rudolph, David Douin



John Burpee, Donald Jenkins, David Douin

John Burpee, Pfaff family, David Douin


THE NATIONAL BOARD

The National Board Remembers

Albert J. Justin

The National Board regrets to announce the April 16 passing of retired Executive Director Albert J. Justin. He was 88 years old.

Mr. Justin was elected executive director of The National Board of Boiler and Pressure Vessel Inspectors on March 12, 1993, following the death of then-executive director Don "Mac" McDonald in 1992. Retiring on March 31, 2001, he was the fifth executive director to head the internationally recognized safety organization since its formation in 1919.

A native of St. Stephen, Minnesota, Mr. Justin began his career in the boiler and pressure vessel industry in 1950 as a boiler operator and assistant chief steam engineer at Deaconess Hospital in Minneapolis. He later joined the Continental Insurance Company, where he was employed for 30 years, the last seven as manager of the loss control department.

In 1984, Mr. Justin joined the state of Minnesota as assistant chief inspector and was appointed chief inspector in 1986 – a position he held until his retirement in February 1993.

As a member of the National Board from 1986 to 1993, Mr. Justin served on numerous task groups. He was chairman of the board from 1989 to 1991 and received the National Board Safety Medal – the organization's highest honor – in 2006.

"The years under Mr. Justin's leadership were among the most energized, if not challenging, periods in the National Board's long and distinguished history," explains National Board Executive Director David Douin. "And while his contributions to our organization are numerous, his efforts on behalf of our industry strengthened it to become the successful entity it is today."

As a National Board commissioned inspector (number 3572), Mr. Justin was active as a member of the Minnesota Boiler Inspectors Association and the American Society of Mechanical Engineers (ASME). Additionally, he served on ASME's Main Committee, the Board of Nuclear Codes and Standards, and as vice chairman of the Boiler and Pressure Vessel Accreditation Committee. He was also a member of the Canadian Standards Association, and an associate member of the American Boiler Manufacturers Association.

A veteran of the U.S. Navy, Mr. Justin attended the Humbolt School of Business. As National Board Executive Director, he served on the board of trustees for the Greater Columbus Safety Council.

PETER DODGE

Manager/Chief Inspector, Province of Nova Scotia

BULLETIN Photograph by Moments in Time Photography Studio

Aunt Agatha had a wonderful assortment of old clocks.

But every time her six-year-old nephew Peter Dodge came to visit, she hid her collection to avoid seeing the timepieces taken apart. By young Peter.

"I love how things work," the Nova Scotia Manager/Chief Inspector admits 50 years later.

By the age of 10, Peter started a small engine repair business. For his twelfth birthday he asked for a gas welding kit to build bigger projects.

"My friends and I spent quite a bit of our youth scouring a nearby junk yard," the Nova Scotia official recalls. "We built everything from scuba gear to go-carts with lawn mower engines."

Born in Truro, Nova Scotia, Peter was one of four Dodge brothers born to John Dodge, a World War II POW survivor and entrepreneur who started an accounting company and later a Mercury auto dealership.

"Growing up in a dealership, I attached myself to top tradesmen with a love of cars. The dealership only added more fuel to my obsession," he observes with a smile.

To prepare his son, the elder Dodge started Peter at the bottom, where the 15-year-old spent the summer washing cars. "The next summer I worked as a mechanic's helper before spending the following two summers learning bodywork from master craftsmen."

As high school graduation approached, the Truro native was faced with making a career decision. "There were only three options: become a mechanic, a machinist, or study engineering." A professor friend of Peter's dad invited him to take a university tour and bestowed upon him advice he remembers to this day: "If you really want something, never, ever quit!"

In 1979, Peter began to study mechanical engineering at Dalhousie University (and later at the Technical University of Nova Scotia). That summer he convinced the Department of Environment that he was a computer programmer, despite having only earned a half credit in programming. "They needed to fix a broken program six years in development. Thinking I bit off more than I could chew, I was fortunate enough to figure out their problem," he remembers.

Peter returned to the university in the fall and spent his free time as a member of the varsity springboard diving team. The next two summers were devoted to working as a machinist at the school's machine shop, where he became involved in a study to determine how to capture energy from waves.

"It was my good fortune to work with several craftsmen who avoided computers in favor of using one's mind," he explains. "It was a wonderfully profound experience that forced me to become more disciplined in my approach to work."

Peter's senior project also exposed him to critical thinking. "I worked with a team to design a wheelchair system that would allow the user to literally transport himself into a modified Ford Escort and operate the vehicle from his wheelchair."

While engineering jobs were plentiful in the years leading up to 1982, the following year marked an economic downturn that frustrated 1983 graduates. The future Nova Scotia official decided early on that rather than waiting around for engineering companies to call, he would make calls to the engineering companies. Cold calls.

On one visit, he struck up a conversation with a man who had parked his antique Chrysler Imperial in front of an engineering company office. A chance conversation with the man – owner of an ASME Section I shop – resulted in a job offer Peter quickly accepted.

"I was hired as a quality control inspector at MBB Mechanical Services. My first assignment was a refining tower constructed in the shop," he explains.

The newly minted mechanical engineer took on a variety of different assignments from designing ASME Section I components to designing and constructing gas D-type package boilers.

"I stayed with the company for 11 years and it was the most gratifying experience of my career," Peter smiles. "It gave me tremendous shop and field experience.

"When the company was purchased I remained for a couple of months as a regional manager before accepting a position in 1995 from then Nova Scotia chief inspector Bob Yeo. I earned my commission within a couple of months of starting the job."

In 1996, Peter assumed responsibility for a fledgling computer program the chief inspector visualized but could not get to work properly. "The program electronically tracks inspections, welder certifications, and power engineering," he explains. Today, this program exists in various forms in several jurisdictions.

In 2006, upon the retirement of Bob Yeo's successor Charles Castle, Peter was named acting chief. In 2008, he was named chief inspector following an interdepartmental competition to determine a replacement.

With a staff of 12 inspectors, the Nova Scotia official now oversees approximately 20,000 objects in the province. He serves on the National Board Committee on Qualifications and possesses a National Board commission with **A** and **B** endorsements.

Now in his mid-50's, the Truro native faces the everyday challenges of life. To stay in shape he begins his day five times a week with a one-mile swim at 5:30 a.m. And although he loves fishing, sports, and being outdoors, "I'm happiest with grease up to my elbows."

Peter and his wife Shelagh have been married for 25 years. Other members of the Dodge family include two sons and two daughters. "Like his dad, my oldest son is in the boiler industry. My youngest son is studying to become a physics teacher while my oldest daughter is on track to become a nutritionist. My youngest daughter hasn't yet found her 'thing'," he explains.

Peter says he still hasn't lost his passion for all things mechanical. "I still get a kick out of working on my cars (a 1994 Miata and 1977 Ford Capri) and fixing things for other people. As a matter of fact, my brothers call occasionally when something around the house needs worked on."

Their only instruction: "Stay away from the clocks!"

The Return of the "**C**" Endorsement Course

BY KIMBERLY MILLER, MANAGER OF TRAINING

Once every few years the National Board conducts its *Authorized Nuclear Inspector Concrete* (**C**) Course. This training is the most specialized course offered by the National Board as it is specifically designed for the inspector assigned to perform inspections during the construction phase of nuclear components, parts, and appurtenances fabricated and assembled in accordance with the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code,* Section III, Division 2.

This five-day course begins by providing students with a fundamental understanding of concrete and concrete construction, then progresses into the duties and responsibilities of the inspector, materials, design, fabrication and construction, construction testing and examination, and structural integrity testing of concrete containments. Although much of this course takes place in a classroom setting, students will also take part in a hands-on concrete workshop in our inspection room. This workshop allows students to work in small groups in order to perform several tests on freshly mixed concrete; for example, air content determination and a slump test.

Although this course is designed for the authorized nuclear inspector looking to receive the "**C**" endorsement, others in the nuclear industry may also find this course of interest. Learners interested in enrolling for the December 7-11, 2015, class may do so via the online enrollment form. This form is accessible from the National Board website's Training menu/Classroom Training Catalog page. Tuition is $1,495.

2015 Classroom Training Courses and Seminars

All training is held at the National Board Training Centers in Columbus, Ohio, unless otherwise noted. Class size is limited and availability subject to change. Check the National Board website for up-to-date availability.

COMMISSION/ENDORSEMENT COURSES

(B/O) **Authorized Inspector Supervisor Course**
TUITION: $1,495, 2.5 CEUs Issued
October 12-16, 2015

(N) **Authorized Nuclear Inspector Course**
TUITION: $1,495, 3.1 CEUs Issued
August 24-28, 2015

(I) **Authorized Nuclear Inservice Inspector Course**
TUITION: $1,495, 2.7 CEUs Issued
September 14-18, 2015

(C) **Authorized Nuclear Inspector (Concrete) Course**
TUITION: $1,495, 2.5 CEUs Issued
December 7-11, 2015

(IC) **Inservice Commission Course**
TUITION: $2,995, 8.7 CEUs Issued
August 10-21, 2015
November 9-20, 2015

(A) **New Construction Commission and Authorized Inspector Course**
TUITION: $2,995, 7.0 CEUs Issued
October 19-30, 2015
November 30 - December 11, 2015

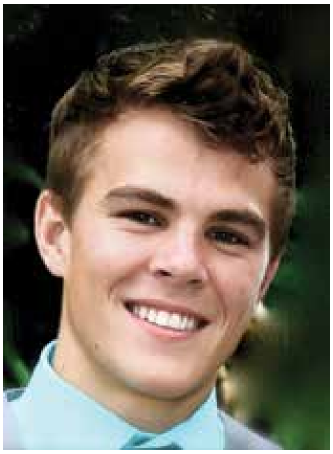
REPAIR SEMINARS

(VR) **Pressure Relief Valve Repair Seminar**
TUITION: $1,495
OFF-SITE TUITION: $1,595
2.6 CEUs Issued
September 28 - October 2, 2015

(RO) **Boiler and Pressure Vessel Repair Seminar**
TUITION: $795
OFF-SITE TUITION: $895
October 13-15, 2015
Charlotte, NC

Three Named 2015-2016 National Board Technical Scholarship Recipients

Three students have been chosen for the 2015-2016 National Board Technical Scholarship award. Mitchell J. Olney of Indiana, Bryant Crouch of Ohio, and Jacob Henningsen of Ohio each will receive $8,000 toward their studies.



Mitchell J. Olney is pursuing a degree in mechanical engineering technology at Indiana Purdue University Fort Wayne (IPFW). He is the son of Commissioned Inspector James Olney. "Growing up I watched my father work as a boiler inspector, making the community a safer place, and I respected him for that. At a young age, I decided I wanted to make a similar impact by becoming an engineer," he says. During his senior year in high school, Mitchell obtained an internship and then summer employment at engineering firm Parco Inc. During his sophomore year at IPFW, he was approached by Play Core, a manufacturer of stadium bleachers, to become an engineer in the CAD design department. "These experiences have been very beneficial, as I have gained confidence in my engineering knowledge and design abilities," he says. Olney made the dean's list in spring 2014 and is president of the IPFW college rugby team. "My ultimate goal is to have a career that has a positive impact on safety with the use of engineering knowledge," he says.

Bryant Crouch is studying mechanical engineering at the University of Cincinnati. He is the son of Commissioned Inspector Robert Crouch. From a young age Bryant was intrigued with engineering. "I decided to shadow my father on many of his field inspections so I could see how equipment in buildings operated and worked together to form a smooth-running mechanical and electrical system," he says. Crouch is part of the university's highly recognized co-op program, and is interning with Heapy Engineering in Dayton, Ohio. During his first co-op term he worked on designing piping and ductwork systems for commercial buildings. "I went on site visits and saw pressure vessels, chillers, and generators, and how they all tie into a mechanical system," he says. "My father opened my eyes to engineering and has taught me to work as hard as I can in everything I do. I fully believe that with hard work and determination, we can truly achieve any goal," he says.

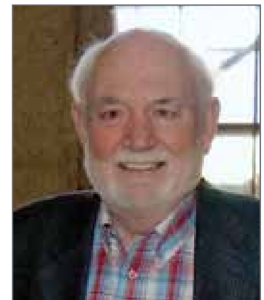
Jacob Henningsen is pursuing a degree in electrical engineering at the University of Cincinnati. He is the son of National Board employee Bill Henningsen. Jacob participates in the university's co-op program. His first co-op was with Integrated Test and Measurement as an engineer technician. "On one project, I travelled to a paper mill where I worked on updating a network of strain measurement gauges on the hanger rods of an eleven-story recovery boiler," he says. He also gained experience working with the research and development team at PowerTap division of Saris Cycling Group, and then with iHeartMedia, where he split time working with two engineers: one who set up and maintained all studio equipment, and another who was in charge of all radio tower equipment. "My internship experiences have given me new perspectives on the ways classroom material can be applied to the real world and my post-graduation career," he says. In addition, Henningsen is the president of the University of Cincinnati Cycling Club.

Member Retirement


William Owens

William Owens, Louisiana chief boiler inspector, retired on February 27, 2015. Mr. Owens served in the US Air Force. In 1980, he joined Hartford Steam Boiler Inspection and Insurance Company in the Los Angeles area as a boiler inspector. In 1982, he worked for the City of Tucson as a boiler machinery inspector, and then returned to Hartford in 1989. In 2000, he joined the State of Louisiana as a boiler inspector. He was promoted to supervisor in 2004, and then to chief inspector six months later. Mr. Owens was a National Board and ASME team leader and held an **A** endorsement.

Sullivan, Surtees, Doty, and Pfaff Remembered


Michael Francis Sullivan

Michael Francis Sullivan, former National Board staff member and independent ASME consultant, passed away February 14, 2015. He was 76 years old. Mr. Sullivan served in the US Air Force from 1960 to 1967. He joined ASME as a volunteer expert in 1987. From 1987 to 1994, Mr. Sullivan worked for the National Board in several capacities, including consultant and manager of international operations. He served on the ASME BPV Committee on Construction of Nuclear Facilities Components and Subgroup on General Requirements. In 1994, he joined ASME as an independent consultant, including serving as senior consultant from 2008 to 2013. In 2011, he received the ASME Dedicated Service Award as testament to his outstanding contributions to the Society.


Nicholas Surtees

Nicholas Surtees, P.E. Eng., and retired Saskatchewan chief inspector, died on February 11, 2015. He was 70 years old. Mr. Surtees was born in Sunderland, Co. Durham, England. He obtained a degree in metallurgy and immigrated to Canada with his wife in 1969. In 1987, he became executive director with the Government of Saskatchewan, Boilers and Pressure Vessels Division. In 2005, he was employed in the Ministry of Justice, Corrections and Policing. He served on the National Board's Board of Trustees and many committees, and received honorary membership following his retirement. As a representative of the Canadian Association of Chief Inspectors, he chaired and served on various committees and was also active with the American Society of Mechanical Engineers.


W.D. "D'Or" Doty

W. D. "D'Or" Doty, PhD, P.E., and former National Board Advisory Committee member, died on March 5, 2015. He was 95 years old. Dr. Doty received his PhD. (metallurgy) from Rensselaer Polytechnic Institute. He published numerous technical articles and co-authored the authoritative book *Weldability of Steels*. He served several terms as a National Board Advisory Committee member representing the welding industry. In 2007, he received the National Board Safety Medal. He was also a Fellow of the American Society for Metals, the American Welding Society, and ASME. Additionally, he was active in a variety of ASME and other industry groups and committees.


Howard D. Pfaff

Howard D. Pfaff, retired South Dakota chief boiler inspector, passed away on February 24, 2015. He was 79 years old. Mr. Pfaff was a 20-year US Navy veteran and served as a chief boiler technician on Navy vessels in Vietnam. He returned to the States in 1969, and spent the later portion of his military career as a Navy instructor at the Great Lakes Training and Recruit Command in Great Lakes, Illinois. His post-Navy experience began with a 14-year career in the insurance industry. He worked nine years as a boiler operator at a steam plant. In 1992, he performed part-time inspection work as a private contractor in South Dakota. In 2000, Mr. Pfaff assumed chief inspector duties as a private contractor.

New Members


Ulrich (Rick) Merkle

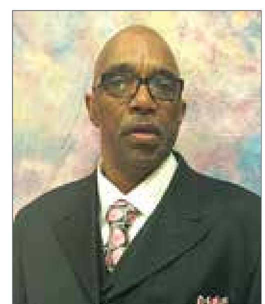

Nathaniel Smith


Aaron M. Lorimor


Trevor S. Seime


Joseph (Donnie) LeSage Jr.


John E. Sharier


Cortney Jackson

Ulrich (Rick) Merkle represents Iowa. Mr. Merkle served in the US Navy from 1979 to 1987, and was a chief boiler technician. His civilian career began at Hartford Steam Boiler Inspection and Insurance Company as a field engineer from 1987 to 1993. In 1993, he worked in Iceland as a thermal heat transfer engineer, and rejoined Hartford as an AI shop inspector in 1997. He joined the State of Wisconsin in 2001 as bureau section chief and remained in that role until joining the State of Iowa in 2014 as interim chief inspector.

Nathaniel Smith represents the Commonwealth of Pennsylvania. Mr. Smith served in the US Navy from 1975 to 1999. He was a master chief boiler technician and served on board five US Navy ships. His civilian career includes a post with the Commonwealth of Pennsylvania from 1999 to 2013 as a commissioned boiler inspector, and then with Zurich Services Corp. from 2013 to 2014 as a risk engineering consultant.

Aaron M. Lorimor represents South Dakota. He was employed as a pipe welder/pipefitter for Zanes Oilfield Service in Utah from 1985 to 1989. The next 18 years he was employed at Dewald Northwest/Wastequip as a welder/lead supervisor. In 2008, he became the welder/project manager/quality control manager at Adams Pipe & Vessel in South Dakota. He was appointed South Dakota deputy boiler inspector in July 2012.

Trevor S. Seime represents North Dakota. Mr. Seime served in the US Navy for over eight years and was a nuclear propulsion machinist mate. His civilian career began as an authorized inspector for Hartford Steam Boiler Inspection and Insurance Company from 1998 to 2006. Next, he became the deputy boiler inspector for the State of North Dakota in 2006, and served in that role until becoming acting chief.

Joseph (Donnie) LeSage Jr. represents Louisiana. Mr. LeSage began his career in 1979 as a pipefitter and draftsman. From 1986 to 1997, he worked for several companies in such roles as welder/fabricator, quality control inspector/design engineer, engineering manager, and estimator. He was employed with Volks Constructors as an autocad draftsman until accepting the position as district chief for the Louisiana State Fire Marshal Office in 2002.

John E. Sharier represents Ohio. Mr. Sharier served in the Army National Guard. He worked as a boiler operator for JII Sales from 1981 to 2004. He became a boiler inspector for the State of Ohio in 2004, and then became boiler inspector supervisor in 2012.

Cortney Jackson represents the City of Detroit, Michigan. Since August of 2000, Mr. Jackson worked for the City of Detroit as a mechanical and boiler inspector before assuming his current role of supervisor, boiler division, for the city. Prior to that, Mr. Jackson was a general welder for the City of Detroit.

National Board *BULLETIN* Index by Title

TRAINING MATTERS:

UPDATES & TRANSITIONS:

National Board *BULLETIN* Index by Author

Code Interpretations

The *National Board Inspection Code* (NBIC) and the American Society of Mechanical Engineers' *Boiler and Pressure Vessel Code* (ASME B&PVC) each issue responses to technical questions submitted by their respective user communities. Interpretations clarify the meaning or intent of existing rules. Section 10 of the NBIC contains an index of all approved interpretations at the time of publishing. A comprehensive index of interpretations is published online at: http://www.nationalboard.org/Index.aspx?pageID=4&ID=22.

The ASME B&PVC Code contains an index of all approved interpretations at the time of publishing, along with the written interpretations for a given date range, at the end of each Section. All written interpretations are also published online at: http://cstools.asme.org/interpretations.cfm.

Following is a selection of interpretation questions currently posted on the respective websites. To review the complete collection of questions, refer to the websites listed above.

NBIC Interpretation

- Interpretation 13-04, Subject: Part 3, 3.3.2 e), (Edition: 2013)
  Question: Is seal welding of inspection opening covers, such as handhole plates or plugs, considered a routine repair?
  Reply: No.

ASME B&PVC Interpretations posted January 2015

- **Section VIII-1**
  Interpretation: VIII-1-98-84E, Subject: UG-16(c) (1998 Edition, 1998 Addenda), Date Issued: October 25, 1999
  Question: May rounding rules provided in ASTM E29 and referred by plate general requirements specifications SA-6 and SA-20 be used when determining compliance with the undertolerance requirements of UG-16(c) in Section VIII, Division 1?
  Reply: Yes. Note: This interpretation originally appeared in Volume 46. In the Question, the phrase immediately preceding "SA-6 and SA-20" has been corrected by Errata to read "requirements specifications."

- **Section IX**
  Interpretation: IX-13-30, Subject: QW-322.1(a), Expiration of Qualification, Date Issued: May 29, 2014
  Background: A welder/welding operator is required to weld with a process within a six-month period, in order to maintain qualification to use that process. A welder/welding operator takes a performance qualification test using a process for which the welder is already qualified (e.g., SMAW), but with different essential variables (e.g., different F-number, progression, etc.). During the performance of the test, the organization responsible for supervising and controlling the test visually examines the weld and determines that it meets the visual acceptance criteria of QW-194. Subsequently, the test coupon is subjected to volumetric NDE or mechanical testing, and fails to meet the acceptance criteria.
  Question: May a failed performance qualification test, utilizing a process for which the welder/welding operator is currently qualified, satisfy the requirements of QW-322.1(a) for maintaining continuity?
  Reply: Yes.

- **Section I**
  Interpretation: I-13-24, Subject: PG-113.1, Master Data Report Form (2013 Edition), Date Issued: June 3, 2014
  Question (1): Are the details for the feed, steam, blowoff, pressure relief valve, manhole and handhole openings required to be shown on Form P-3A, Engineering-Contractor Data Report for a Complete Boiler Unit?
  Reply (1): No.
  Question (2): When using Form P-3A, Engineering-Contractor Data Report for a Complete Boiler Unit, should the appropriate Manufacturer's Data Report Forms be used to document the details for the feed, steam, blowoff, pressure relief valve, manhole and handhole openings?
  Reply (2): Yes.

- **Section IV**
  Interpretation: IV-13-12, Subject: HC-213, Workmanship, Finish, and Repair, Date Issued: January 23, 2014
  Question: Under Part HC of Section IV, specifically HC-213(a) and (b), is welding for esthetic repairs (not for areas where there is leakage and structural reinforcement) permitted?
  Reply: No.

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