Docket No.: R.23-01-007
Date: June 30, 2023
Commissioner: Douglas
ALJ: Seybert
Witness: Samuel Miranda

BEFORE THE PUBLIC UTILITIES COMMISSION OF THE STATE OF CALIFORNIA

Implementing Senate Bill 846 Concerning Potential Extension of Diablo Canyon Power Plant Operations	R.23-01-007 (Filed January 14, 2023)

OPENING TESTIMONY OF SAMUEL MIRDANDA ON BEHALF OF SAN LUIS OBISPO MOTHERS FOR PEACE ON PHASE 1 TRACK 2 ISSUES

Samuel Miranda on behalf of SLOMFP
c/o
Sabrina Venskus
Venskus & Assocaites, A.P.C.
603 West Ojai Avenue, Suite F
Ojai, CA 93023
Phone: 805.272.8628
Email: venskus@lawsv.com

VERIFICATION

The statements in the foregoing document are true and correct to the best of my knowledge. The facts presented in the forgoing document are true and correct to the best of my knowledge, and the opinions expressed therein are based on my best professional judgment.

I declare under penalty of perjury under the laws of the state of California that the foregoing is true and correct. Executed on June 30, 2023, in Leesbur
g, Virginia.

Samuel Miranda

Samuel Miranda

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ATTACHMENTS

Attachment A - Resume

I. STATEMENT OF QUALIFICATIONS

Q. What is your professional experience relating to your testimony in this proceeding.

A. Please see a copy of my resume attached as Attachment A.

Q. What is the subject of your testimony?

A. To provide testimony on questions Related to Pub. Util. Code Section 712.8(c)(2)(B).

II. OPINIONS AND CONCLUSIONS

A. The types of activities the U.S. Nuclear Regulatory Commission (NRC) might include as potential conditions of license renewal.

1. Background.

The NRC issues operating licenses for 40 years, under 10 Code of Federal Regulations ("CFR") 50, and may renew them, for up to an additional 20 years, under 10 CFR 54. That is, operating licenses that are "renewed" are conditioned upon compliance with the aging management and environmental requirements of 10 CFR 54, and upon continued compliance with the safety requirements of 10 CFR 50 (i.e., with the plant's current licensing basis ["CLB"]), for the extended period of operation.

The NRC has four fundamental requirements for renewal of a nuclear reactor license:

- The plant must continue to meet all applicable safety requirements, as set forth in the NRC's regulations. The NRC will review the plant's safety record and its plans for managing the effects of aging on the plant's structure, systems, and components.
- The licensee must develop plans for managing the effects of aging on the plant's safety. These plans must address the potential for aging to lead to failures or malfunctions of plant equipment and must include measures to mitigate these risks.
- The licensee must address the potential environmental impacts of continued operation, including the potential for continued operation to lead to releases of radioactive material to the environment and must include measures to minimize these releases.

- The licensee must demonstrate that it will be able to continue to operate safely and comply with all applicable federal regulations, including adequate financial resources to maintain the plant in good condition, to pay for its operating costs, and to meet all its financial obligations.

The NRC may also impose additional conditions on a license renewal application, depending on the specific circumstances of the plant. The NRC's guidance for implementation of 10 CFR 54 is outlined in the Generic Aging Lessons Learned (GALL) report (NUREG-1801) and the Standard Review Plan for License Renewal (NUREG-1800). There is also Regulatory Guide 1.188, which specifies the format and content of the safety aspects of a license renewal application ("LRA"). Further information regarding the license renewal process, and the potential for imposition of any conditions for renewal is available in references [1] [2] and [3]. The NRC intends to revise these guidelines as generic renewal issues are resolved.

2. **Potential license conditions – general and deferred maintenance.**

Potential conditions of license renewal, for a specific plant, like DCPP, are considered by members of the NRC staff during their reviews of the licensee's LRA. Generally, the NRC staff and licensees endeavor to avoid the imposition of conditions of license renewal by resolving issues during the LRA reviews. If negotiations are unsuccessful, the NRC Staff may impose license conditions. The results of the NRC Staff's review, including negotiated upgrades and license conditions, are reported in the safety evaluation report ("SER") issued at the conclusion of its LRA review.

Because PG&E has not yet submitted a license renewal application, it is premature to speculate what deficiencies the NRC Staff may find and require PG&E to address through negotiated upgrades or license conditions. However, it is reasonable to anticipate that the NRC staff will require PG&E to address a number of safety issues during the license renewal term that PG&E would have resolved during the initial license term if it had not been preparing for shutdown and decommissioning of DCPP. These activities would have to be addressed (i.e., resolved or completed) before any period of extended operation (e.g., for five or 20 years) could be approved.

Any new requirements (e.g., requirements that have not already been identified) could be subject to the Backfit Rule (10 CFR 50.109), which requires the NRC to justify their imposition with a cost/benefit analysis. The analysis would likely be highly subjective and susceptible to objections from the licensee and other stakeholders. Consequently, an acceptable Backfit Rule analysis could require years of effort and negotiation. Neither regulator nor licensee would readily enter into such a process.

3. **Potential license conditions – environmental.**

Pursuant to 10 CFR 50.36, which requires technical specifications on effluents from nuclear power reactors, the NRC Staff may impose some additional conditions on the CLB to protect the environment. Changes in the environment, and in environmental standards and requirements, occurring over 40 to 60 years, could lead to some unforeseen license conditions. For example, once-through cooling might be considered to be no longer acceptable, or new roads or pipelines might be added, or aircraft flight paths could change.

The NRC declares that the safety requirements of the CLB will be satisfied throughout the extended period of operations [13] and then requires licensees only to establish and follow aging management programs. It is expected that continual improvements and upgrades would be required during extended operations to maintain compliance with CLB requirements. This would be analogous to safety requirements that are applied to very old automobiles. If they are to be operated on public streets, they must have should belts, head restraints, anti-lock brakes, catalytic converters, the ability to burn unleaded gasoline. Old automobiles that don't meet all the requirements are designated as antique vehicles that are restricted to certain periods of operation. The NRC does not impose any comparable limits upon nuclear plants that are to operate with "renewed" licenses.

4. **Potential license conditions – insurance and financial security.**

License conditions could also result from PG&E's continuing responsibility to maintain and report the insurance or financial security it holds to cover the costs that would be necessary to operate, maintain, and stabilize the reactor and decontaminate it and the reactor station site. In this respect, maintenance could include the cost of improvements and updates to hardware, software, training, and procedures that would be very hard to predict over the period of extended operation.

5. **Potential license conditions that are required to maintain the current licensing basis (CLB).**

The NRC requires that, "the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term." [13]

The "plant-specific licensing basis" or CLB is defined as:

> "the set of NRC requirements applicable to a specific plant and a licensee's written commitments for assuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications

> and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR Parts *2,* 19, 20, 21, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100, and appendices thereto; orders; license conditions, exemptions; and technical specifications. It also includes the plant-specific design-basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in such docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations, or as described in licensee event reports." [1]

10 CFR 54.35, Requirements during term of renewed license, states that, "During the term of a renewed license, licensees shall be subject to and shall continue to comply with all Commission regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 52, 54, 55, 70, 72, 73, and 100, and the appendices to these parts that are applicable to holders of operating licenses or combined licenses, respectively."

Chapter 15 of the Final Safety Analysis Reports (FSARs), in the CLB, presents accident analyses that are grouped according to postulated accidents' expected frequencies of occurrence and the severity of their predicted consequences. Accidents that are expected to have a high frequency of occurrence must be shown to produce anodyne consequences. Accidents that pose the greatest danger to the public must be limited to very low expected frequencies of occurrence. These nuclear safety criteria were defined, almost half a century ago, in an American Nuclear Society (ANS) Standard [5] and reiterated by the NRC. [6] They are cited and applied in the Diablo Canyon FSAR.

One class of accidents, Condition III, or Infrequent Incidents, are defined as "Incidents, any one of which may occur during the lifetime of a particular plant." These events could result in the failure of not more than a small fraction of the fuel rods, and the resultant release of radioactivity must be limited to levels that would not interrupt or restrict public use of areas outside the exclusion radius. The expected frequency of Condition III events is about once in a plant's operating lifetime. In the CLB, that would be 40 years.

Therefore, it follows that a significant extension of the operating lifetime by 20 years (i.e., by 50%) could result in a significant increase in the frequency of "Infrequent Incidents." This implies that a design improvement or a plant operational modification might be required, to maintain the established once-in-a-lifetime expected frequency. That is, a 20-year extension in a plant's operating lifetime could necessitate a change to decrease the expected frequency of occurrence of Infrequent Incidents from ≤ 1 in 40 reactor-years of operation to ≤ 1 in 60 reactor-years. Currently, this is not an NRC condition of license renewal. If it were to become a condition of license renewal, it could require a costly improvement in plant design or operations before the renewal could be approved. In this case, the modifications could be as simple as a probabilistic analysis to support a determination that no change is necessary, or the revision of some CLB accident analyses, or the addition or modification in operating procedures, or possibly some expensive plant design and equipment changes.

If the NRC does not impose a license condition of this sort, the consequences could be seen in an increased frequency of preventable accidents, which can cause very costly damage. That is, an Infrequent Incident, if not mitigated properly, could effectively end a plant's operating lifetime. Recall that the accident at Three Mile Island Unit 2 involved an Infrequent Incident (i.e., a stuck open pressurizer relief valve), which was aggravated by several profound operator errors. The result was a partial core meltdown. At the time of this Infrequent Incident (1979), Three Mile Island, Unit 2, had been operating less than one year.

6. **Potential license conditions related to aging equipment.**

Other conditions could arise, as the NRC reviews systems, structures, and components according to 10 CFR 54.21 to ascertain whether they will continue to perform their intended functions for the period of extended operation. The review will include relevant time-limited aging analyses to determine whether these systems, structures, and components will continue to perform their intended functions for the period of extended operation.

Examples of aging issues and components that may need to be addressed:

i. Pressure vessel welds. Pressure retaining welds can be susceptible to pressurized thermal shock (PTS). PTS can occur whenever colder water is introduced into the reactor coolant system (RCS) or vessel (RPV) at high pressures. For example, the emergency core cooling system (ECCS), when actuated by a spurious signal or an operator error, would put relatively cold water into the coolant pipes and into the reactor vessel without depressurization of the RCS. [7] The colder water effect can be aggravated by poor or incomplete mixing ECCS water with reactor coolant, especially near the ECCS entry points and RPV nozzles. As the plant is operated and ages, small cracks can develop in these welds. The cracks can get larger as cold-water cycles occur. Eventually, they can become large enough to jeopardize the integrity of a weld if it undergoes another cold-water cycle. If all the pressure retaining welds, at the RPV intake nozzles, are of similar composition, then

PTS-induced through-cracking could occur at all the nozzles. Breaks at more than one location, in the RCS, are not reported in Chapter 15 of the FSAR, since they are not analyzed.

ii. Usage factors. The licensed operating lifetime of a nuclear power plant can be shortened by exceeding the number of allowable design cycles (i.e., temperature and pressure cycles that occur as the result of normal operations, such as heat up and cooldown), or by accidents, or by natural causes (e.g., reaching reactor vessel embrittlement limits.) For example, an accident at Three Mile Island ended its operating lifetime only three months after it began. As another example, reactor vessel embrittlement contributed to Yankee Atomic's decision to end the operating lifetime of its Rowe plant.

Design cycles can be counted and compared to these limits. The cumulative usage factor ("CUF") is the number of counted cycles divided by the number of allowed cycles for each component in the plant. A CUF of 1.0 for a component means the CUF is fully depleted (i.e., its safety margin is zero). Components with lower CUF values have some lifetime remaining. For example, a plant may be designed to tolerate 400 reactor trips (i.e., emergency shutdowns or scrams) from full power during its lifetime. Conceptually, ten trips per year, in such a plant would deplete its reactor trip CUF by the end of its 40-year lifetime. In practice, a plant that experiences more than a couple of reactor trips in any one year would trigger some close attention from the NRC. It could lead to a license condition to reduce the number of unnecessary reactor trips per year. The NRC regards unnecessary reactor trips as a potential safety issue (i.e., too many challenges to safety systems).

iii. Steam Generators. Steam generator tubes are subject to corrosion and age. When they leak, they are plugged, usually during a scheduled outage. Plugged tubes decrease the available heat transfer area and consequently, the plant efficiency. Too many plugged tubes eventually require replacement of the steam generators. Operating experience indicates that steam generator lifetime is generally not as long as 40 years.

B. Potential upgrades, and associated costs or cost ranges, that might be needed to address deferred maintenance and NRC's potential conditions of license renewal at Diablo Canyon.

1. Steam generator replacement.

The steam generators at the Diablo Canyon units are at least 15 years old.

> "Overall, the prospect for continued operation of PWR's in the United States is good, but the prospect for long-term operation of original steam generators with Inconel 600 mill-annealed tubing is poor. Steam generator problems rank second, behind refueling outages, as the most significant contributor to lost electricity generation. The only exceptions are likely to be those reactors that recently began operation, where the lessons learned in such areas as water chemistry, tubing material, tube support plate material, and tube support plate design and attachment were incorporated from the very beginning of unit operation." [8]

Replacement steam generators are improved and expected to yield better performance than the originals; but their lifetime is not yet known. It is likely that the DCPP steam generators will have to be replaced during a lifetime extension of as much as 20 years (i.e., it is not likely they last as long as 35 years). For example, consider the steam generators at Watts Bar Units 1 and 2, near Spring City, Tennessee. Watts Bar Unit 1 began operation in 1996. Its steam generators were replaced 10 years later, in 2006. Watts Bar Unit 2 began operation in 2016. Its steam generators were replaced only 6 years later, in 2022. [14]

2. Deferred maintenance can raise the cost of equipment repair and replacement.

Deferred maintenance costs can be higher than the costs of normal (scheduled) maintenance, since lead times for the delivery and replacement of equipment could be shorter, and the costs of delays could be greater (i.e., there could be less margin for delay).

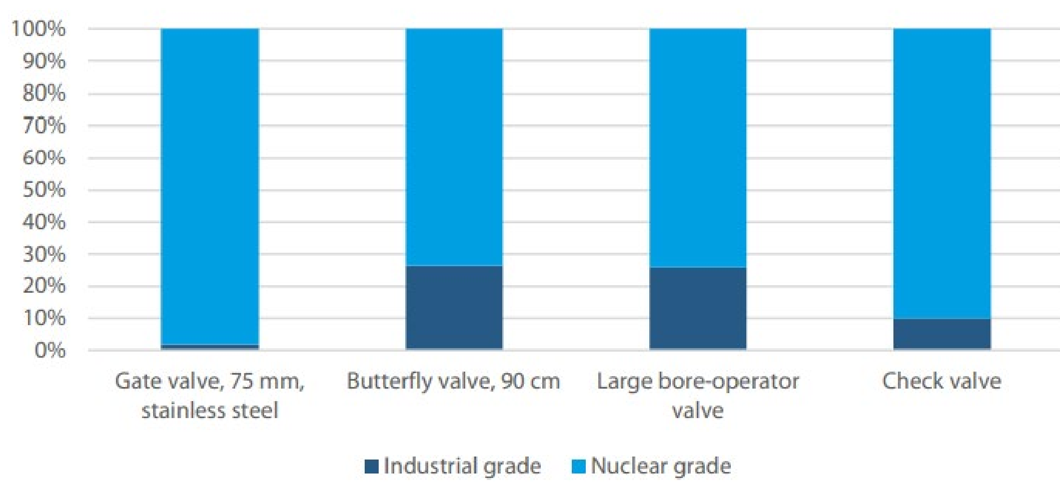
Equipment that is active, and in constant use (for example, charging pumps) needs to be repaired or replaced periodically. Operators of old plants like DCPP are motivated to obtain exact replacements from the original designers and manufacturers, even when improved, more efficient models become available. Sometimes, the original designers and manufacturers are no longer in business, or they no longer offer the original models. Licensees who wish to replace worn or obsolete equipment with anything that is different must justify differences in performance and reliability with the NRC. The process would likely require new analyses and evaluations of performance and response characteristics in accident situations. These can be expensive and time-consuming. NRC review and approval of new analyses can also be expensive and time consuming. The associated costs are generally expected to increase as plants age. The costs can be especially high in situations wherein maintenance has been deferred, and delays and lead times must be limited.

Plant components require extensive testing and verification to ensure they can reliably function under accident conditions. This can require detailed documentation of every component at each step of its manufacture. This level of documentation can be expensive and time-consuming. For example, here is an engineer's description of how QA requirements were met during the construction of the Diablo Canyon nuclear plant:

> "Simple field changes to avoid physical interference between components (which would be made in a conventional plant in the normal course of work) had to be documented as an interference, referred to the engineer for evaluation, prepared on a drawing, approved, and then released to the field before the change could be made. Furthermore, the conflict had to be tagged, identified and records maintained during the change process. These change processes took time (days or weeks) and there were thousands of them. In the interim the construction crew must move off of this piece of work, set up on another and then move back and set up on the original piece of work again when the nonconformance was resolved… Every foot of nuclear safety-related wire purchase is accounted for and its exact location in the plant is recorded. For each circuit we can tell you what kind of wire was used, the names of the installing crew, the reel from which it came, the manufacturing test, and production history." [9]

Similar documentation requirements apply to the manufacture of nuclear-grade components. A former engineer describes the process: "Many moons ago I did (design and manufacturing) for a company that made both (section VIII and section III (nuclear) vessels) and my memory is that it was essentially the same design work with much more documentation and paperwork required for the 'N' stamp vessel. When the paper weighed about what the vessel did, it was ready to ship." [10] An EPRI analysis found that nuclear-grade components could be fifty times more expensive than off-the-shelf industrial grade ones. [9]

Figure 28: **Cost gap between nuclear- and industrial-grade valves**



Nuclear-grade components don’t necessarily have higher performance requirements than conventional components. The additional cost often stems from extensive documentation and rigorous testing requirements. These requirements also lead to higher costs by damping market competition among manufacturers. Some manufacturers avoid these requirements by leaving the nuclear-grade components market.

3. Annealing.

Reactor pressure vessel embrittlement is becoming an increasingly important issue as plants continue to age, and their components (i.e., reactor vessels) are subjected to neutron irradiation. To ensure acceptable levels of vessel integrity and plant operability, vessel embrittlement must be maintained within NRC regulatory limits. Annealing can restore some material toughness, for some time. Annealing is performed in place by heating the reactor vessel walls to a temperature sufficient to remove some or all neutron damage effects. Although annealing has not been performed at commercial plants in the United States, it has been used successfully to remove radiation effects from reactor vessels at several Soviet-built reactors.

It seems to be feasible to perform an in-situ thermal anneal of axial and circumferential vessel welds. Shielding the reactor internals during the annealing process is also possible. However, personnel radiation exposure can increase during the annealing operation, and the removal and reinstallation of the closure head assembly, the reactor vessel internals, and all fuel assemblies. Plant personnel exposure can be more than 55 rem. [11]

Operating lifetime might be limited, if nuclear reactor pressure vessel welds become brittle, or cracks develop and lengthen, and annealing is not undertaken. For example, embrittlement was major factor in Yankee Atomic Electric Company’s decision to decommission its Yankee-Rowe power plant outside Boston, in 1992, eight years before its license was due to expire. Ultimately, the decision to anneal a vessel would likely depend upon economic and political factors, as well as underlying embrittlement effects.

4. Financial security and other economic factors.

PG&E is required to maintain and report its insurance and financial resources that it can allocate to the costs of operation, maintenance, and reactor stabilization and decontamination (including the reactor site). The requirement, which is in the CLB, would remain in effect during any authorized extended period of operation.

Economic factors are the major contributors to nuclear reactor shutdowns [12] at any time in plant life. They should be weighed as major factors in decisions to begin continued operations beyond nuclear plants' design operating lifetimes.

Table 1. U.S. Nuclear Reactor Shutdowns: 2013-2021

Organized by Shutdown Date

Reactor	State	Shutdow	Generatin g Capacity	Start-	Major Factor(s)
Crystal River 3	Florida (FL-11)	Feb. 2013	86	1977	Cost of major
Kewaunee	Wisconsin (WI-8)	May 2013	56	1974	Operating losses
San Onofre 2	California (CA-49)	June 2013	1,070	1983	Cost of replacing
San Onofre 3	California (CA-49)	June 2013	1,080	1984	Cost of replacing
Vermont	Vermont (VT-at	Dec. 2014	62	1972	Operating losses
Fort Calhoun	Nebraska (NE-1)	Oct. 2016	47	1973	Operating losses
Oyster Creek	New Jersey (NJ-3)	Sept. 2018	61 4	1969	Agreement with state to avoid
Pilgrim	Massachusetts	May 2019	68	1972	Operating losses;
Three Mile	Pennsylvania	Oct. 2019	80	1974	Operating losses
Indian Point 2	New York (NY-	April 2020	1,020	1974	Low electricity
Duane Arnold	Iowa (IA-1)	Aug. 2020	60	1975	Lower-cost
Indian Point 3	New York (NY-	April 2021	1,038	1976	Low electricity
		Total	**9,436**		

Source: CRS, with information from the U.S. Energy Information Administration and plant operator announcements.

Notes: Generating capacity numbers reflect "Net Summer" capacity.

The costs of maintenance, over a period of extended operation, should include improvements and updates to hardware, software, training, and procedures that could be required by regulations, market forces, and environmental requirements. Some improvements and updates could be required for example, to make the "renewed" plants consistent with newer, safer plants, any such changes would be expected to be constrained by cost and other economic factors.

C. The costs associated with deferred maintenance or NRC conditions of license renewal at Diablo Canyon which are "too high to justify."

1. Steam generator replacement.

If replacement of the steam generators is necessary during the plant lifetime extension, then the cost could be too high to justify, especially if the replacement is necessary soon after the extended period commences. Poor planning and the design of the steam generator replacements could effectively end the lifetime of the DCPP plants at any time (e.g., the San Onofre experience with defective replacement steam generators).

2. Annealing.

Annealing can be effective in restoring ductility, for a time, at high cost, and with some radiation exposure to plant personnel. Yankee Rowe considered annealing and decided, instead, to decommission its Yankee-Rowe plant eight years before its license expiration. If annealing becomes necessary to continue operations, then that cost could be too high to justify.

Q. Does this conclude your Opening Testimony?

A. Yes it does.

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[13] Federal Register / Vol. 60, No. 88 / Monday, May 8, 1995 / Rules and Regulations, page 22461

[14] Nuclear Engineering International, “Swapping steam generators at Watts Bar”, 5 April 2023 https://www.neimagazine.com/features/featureswapping-steam-generators-at-watts-bar-10737329/

Attachment A

<u>SAMUEL MIRANDA CURRICULUM VITAE</u>

I hold BS and MS degrees in nuclear engineering from Columbia University, and a PE license in mechanical engineering, in the Commonwealth of Pennsylvania. (I also hold master's degrees in business and public administration, and in international affairs from the University of Pittsburgh.)

I have more than four decades of experience in reactor safety analysis and licensing: at Westinghouse, the NRC, and as an independent contractor.

At Westinghouse (25 years), I worked in their Nuclear Safety Department, where I performed nuclear safety analyses of Westinghouse plants, and other PWR plants, including Soviet-designed plants to resolve reactor safety questions, to improve nuclear power plant operability, and to support the licensing of nuclear plant modifications, core reloads, and changes in operating procedures. I also developed standards and methods for use in nuclear safety analysis, and automatic reactor protection systems design. My work in reactor protection systems design included the preparation of functional requirements, component sizing, and determination of setpoints, time response limits, and Technical Specification revisions. In the 1980s, I managed a program, for more than 30 utilities in the Westinghouse Owners Group, to develop and patent a system to improve power plant availability and safety by reducing the frequency of unnecessary automatic reactor trips. The system was approved by the NRC and required as a condition of restart for at least one plant.

As an independent contractor, I worked at the Salem plants, where I prepared a license amendment request, and its associated Technical Specification revisions, to win NRC approval of an upgrade of the plants' power-operated relief valves to safety-grade status, with water relief capability (the industry's first). I also consulted with the Brazilian navy, in São Paulo, in a review of the automatic protection system logic and performance requirements for a proposed submarine propulsion reactor design.

At the NRC (14 years), I worked in the Office of Nuclear Reactor Regulation, where I reviewed license amendment requests for license renewals, power upratings, and modifications of protection systems in PWR and BWR reactor systems. This included presenting and defending review results before the Advisory Committee on Reactor Safeguards (ACRS). I also revised several sections of the Standard Review Plan (NUREG-0800) and presented the revisions to the ACRS. Before I retired, in mid-2014, I reviewed and approved the reactor safety systems of the Navy's new Ford class of aircraft carriers. (I was recognized with a letter of commendation from the Navy's Chief of Nuclear Operations.)

Sam Miranda holds BS and MS degrees in nuclear engineering from Columbia University, and a PE license in mechanical engineering, in the Commonwealth of Pennsylvania. (He also holds master's degrees in business and public administration, and in international affairs from the University of Pittsburgh.)

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