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Docket: NRC-2014-0137
Technical Basis for Regulatory Guidance on the Alternate PTS Rule

Comment On: NRC-2014-0137-0001
Draft Guidance Regarding the Alternate Pressurized Thermal Shock Rule

Document: NRC-2014-0137-DRAFT-0012
Comment on FR Doc # 2015-05754

Submitter Information

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80 FR 13449

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General Comment

Dear NRC,

Please see the attached files:

1. June 1983 Popular Science article, "Thermal shock--new nuclear-reactor safety hazard?", by Edward Edelson.

2. January 27, 1970, Advisory Committee on Reactor Safeguards (ACRS), chaired by Joseph M. Hendrie, "REPORT ON PALISADES PLANT," sent to AEC Chairman Glenn T. Seaborg.

3. MEMORANDUM and ORDER (Ruling on Petition to Intervene and Request for a Hearing), NRC Atomic Safety and Licensing Board Panel, In the Matter of: ENTERGY NUCLEAR OPERATIONS, INC. (Palisades Nuclear Plant), LBP-15-17, Docket No. 50-255-LA, ASLBP No. 15-936-03-LA-BD01, May 8, 2015.

In Edelson's Popular Science article (at Page 3 to 4 of 7 on PDF counter), Theodore U. Marston of the Electric Power Research Institute (EPRI) in Palo Alto, CA admits that used car frames were used to fabricate early RPVs. He stated "We used a lot of auto stock...When you melt it, you can't get all the wiring out."

Beyond Nuclear wonders if Palisades was one of these early RPVs to use used cars in its fabrication?

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E-RIDS = ADM-03
Add = G. Stevens (gls4) - M. Kirk (mtk)
S. Burton (sxb3)

The concern is the introduction of uncontrolled, and unknown, amounts of soft metals, such as copper, manganese, nickel, and phosphorous, as impurities into the metallurgical mix of RPV walls and welds.

10CFR50.61a does not conservatively address the potential for uncontrolled, unknown amounts of soft metal impurities in the welds, plates, and forgings of RPVs, especially old ones like Palisades. But these are the very "weak links in the chain" that are vulnerable to neutron radiation bombardment over time, the cause of RPV embrittlement.

To make matters worse at Palisades, as communicated by ACRS Chairman Hendrie to AEC Chairman Seaborg on Jan. 27, 1970, "the omission of the thermal shield" is an unfortunate "feature" of the Palisades reactor. As our environmental coalition's expert witness, Arnie Gundersen, Chief Engineer at Fairewinds Associates, Inc. pointed out in his Expert Declaration (also previously submitted as a public comment in this DG-1299/NUREG-2163 proceeding) filed on Dec. 1, 2014 in our intervention against Entergy Nuclear's LAR for 50.61a regulatory relief, the inclusion of the thermal shield likely would have prevented the ever more dangerous embrittlement of the Palisades' RPV from the get go.

Hendrie also assured in 1970 that "Surveillance specimens in the vessel will be used to monitor the radiation damage during the life of the plant. If these specimens reveal changes that affect the safety of the plant, the reactor vessel will be annealed to reduce radiation damage effects. The results of annealing will be confirmed by tests on additional surveillance specimens provided for this purpose...".

Of course, these ACRS, AEC (NRC) promises have been broken. These assurances were false.

A surveillance specimen withdrawn from the Palisades RPV in the early 1980s was simply declared by Palisades' owner/operator (Consumers Power at that time) to be too accidentally over-irradiated to be of any use as a PTS risk/RPV embrittlement test sample. For its part, NRC has said that that particular capsule (A-60) was not needed, as another capsule, located directly across the RPV, could serve as a proxy. These arguments were summarized in the attached May 8, 2015 NRC ASLBP MEMORANDUM and ORDER (Ruling on Petition to Intervene and Request for a Hearing) (see pages 30-34, or 32-36 of 51 on PDF counter).

However, Arnie Gundersen also pointed out in his Dec. 1, 2014 expert declaration that the neutron flux (and thus fluence) at various points on the Palisades' RPV differed significantly -- such as even at points diametrically opposed to each other on the RPV circumference.

Also, an unexplained disconnect in Entergy's logic is how Consumers Power could have known capsule A-60 was accidentally over-irradiated, and not usable, as a PTS risk/embrittlement test sample, if it was not tested?

The environmental coalition has protested, and continues to protest, the exclusion of A-60's data from Palisades' RPV embrittlement analysis, in addition to Entergy's refusal to pull and test another capsule until 2019 (and then its refusal, apparently, to pull and test any more, of the three remaining capsules, thereafter).

The relevance to this DG-1299/NUREG-2163 proceeding is that we also protest the broad aspects of this regulatory guide and its technical background, part and parcel of 10CFR50.61a, that allow for the exclusion of data, and ignoring of readily available physical surveillance specimens, as has taken place and is still taking place at Palisades.

Of course, NRC has never required Palisades' dangerously embrittled RPV to be annealed. For this reason, we protest DG-1299/NUREG-2163's permissive provisions which envision allowing operations for another 16 years, despite (false, broken) promises made 45 years ago.

Thank you for considering these public comments.

Sincerely,

Kevin Kamps, Beyond Nuclear (and board member, Don't Waste MI, representing the Kalamazoo chapter)

Attachments

June 1983 Popular Science

1 27 1970 ACRS to AEC Hendrie to Seaborg

5 8 15 ASLB Ruling on Petition May 8, 2015-1

Popular Science

THERMAL SHOCK
-new nuclear-reactor safety hazard?

5 WILD WINDMILLS

The new generation of incredible do-it-all hand-held computers

- Preview: '84 cars
- Component-TV explosion
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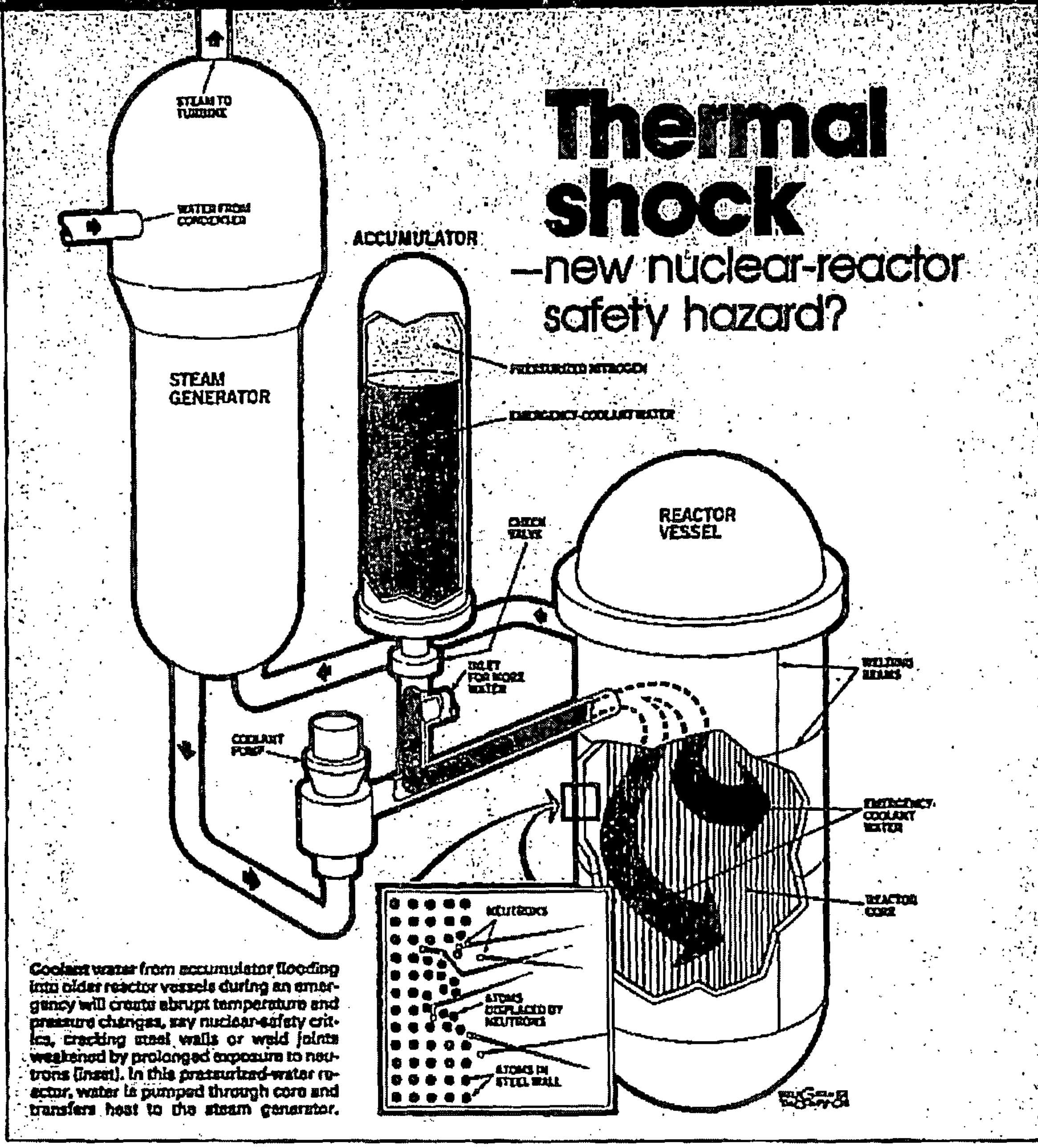
Thermal shock

—new nuclear-reactor safety hazard?



Coolant water from accumulator flooding into older reactor vessels during an emergency will create abrupt temperature and pressure changes, say nuclear-safety critics, cracking steel walls or weld joints weakened by prolonged exposure to neutrons (inset). In this pressurized-water reactor, water is pumped through core and transfers heat to the steam generator.

Could cooling water rupture brittle reactor walls? Here are the facts

By EDWARD EDELSON
DRAWING BY EUGENE THOMPSON

"There is a high, increasing likelihood that someday soon, during a seemingly minor malfunction at any of a dozen or more nuclear plants around the United States, the steel vessel that houses the radioactive core is going to crack like a piece of glass. The result will be a core meltdown, the most serious kind of accident, which will injure many people, destroy the plant, and probably destroy the nuclear industry with it."—Demetrios L. Basdekas, *The New York Times*, March 29, 1982.

Basdekas, a reactor-safety engineer with the Nuclear Regulatory Commission, continued his article to warn that radiation is making the metal reactor vessels at some nuclear plants brittle. As a result, he wrote, water used to flood and cool reactor cores in an emergency could cause a meltdown instead of preventing one. The cause: abrupt changes in reactor pressure and temperature—a condition called pressurized thermal shock—would crack brittle vessels, allowing emergency water to escape.

The safety engineer's "piece-of-glass" charge quickly focused attention on thermal shock:

- The NRC commissioners held a public meeting.
- Rep. Ed Markey of Massachusetts called a congressional hearing.

Continued

● Work on what was supposed to be a definitive study of the thermal-shock issue was accelerated by the NRC.

And the kind of debate that has become quite familiar in recent years has predictably erupted. Electrical utilities, reactor manufacturers, and the Nuclear Regulatory Commission say that the pressurized-thermal-shock problem is well in hand and that the "piece-of-glass" charge is absurd. Critics say that the nuclear people are talking through their hats because there simply isn't enough information available to assess the danger of pressurized thermal shock.

I've recently talked to experts on both sides of the question. At the moment there are no pat answers. But information about the hazard of thermal shock is accumulating steadily. Here is what you need to know.

Pressurized thermal shock has been widely publicized only recently. But inklings of a problem emerged in the 1960s.

At one power-plant reactor, a worker peered into a video monitor and manipulated a robotic arm down into the radioactive water of a 40-foot-high reactor vessel. He slowly fished out a small basket hanging near the thick metal wall of the reactor. Inside the basket was a jumble of pencil-size steel bars, each alloyed with various metals and each bearing a V-shaped notch.

At a nearby test area, he carefully unloaded his irradiated catch behind shielded-glass windows. Deft maneuvers with another robotic arm positioned each steel bar under a wedge-shaped hammer. Then, as samples were cooled or heated, he pushed a button, and the hammer slammed into the notches.

This routine Charpy test (named for its developer) yielded expected results: At lower temperatures, where metals become brittle, samples broke easily. Higher temperatures—like those in your kitchen oven—made the steel more ductile. Heated steel samples absorbed more hammer energy before snapping.

But something unexpected occurred when the worker slammed his test hammer onto bars alloyed with tiny amounts of copper. The steel—even warmed—broke easily. He raised the temperature. Still the brittle bars snapped. Finally at about 300 degrees F, the bars became ductile instead of brittle. The presence of copper seemed to be producing strange results. Soon workers at other power and research reactors discovered the same unexpected embrittlement.

What puzzled everyone was the speedup of embrittlement because of the presence of copper, not the results of the standard Charpy tests on exposed metal samples. This technique—gradually changing metal temperatures and measuring how much hammer energy the metal can absorb without breaking—actually tests radiation damage. Radiation tends to make all metals brittle; irradiated metal must be raised to a higher temperature before it will become ductile. This shift in the transition temperature from brittle to ductile is a measure of radiation damage.

Nuclear researchers, aware of metal embrittlement, had earlier exposed samples to intense radiation. But the surge of reactor construction beginning in the 1960s found engineers without enough reliable data. To an-

> **"Copper was used to prevent rust. Someone probably got a prize for the suggestion"**

swer questions about long-term radiation effects on metal, baskets of Charpy samples had been positioned in early reactors.

The principal cause of embrittlement was known to be neutrons, the atomic particles emitted by nuclear fission in the reactor core, colliding with metal in the reactor. "It's like billiards," says one expert. "Although metal atoms are much heavier than neutrons, when a high-energy neutron collides with a metal atom, the neutron forces the atom from its lattice—the geometric array of atoms."

The Charpy tests of the 1960s revealed that just a little copper in a steel alloy hastens embrittlement. Since that time, though, researchers have been uncertain why the presence of copper hastens radiation damage. Theodore U. Marston, who works on thermal shock at the Electric Power Research Institute in Palo Alto, Calif., says there's now strong evidence that neutron bombardment makes the copper clump together.

"Copper starts out in a solid as atoms fairly evenly distributed. Under radiation the atoms tend to come together as copper particles," he said. New instruments that let researchers see atoms within metals show this clumping effect, Marston says.

As the first discoveries of brittle irradiated steel containing copper became known, anxiety began to spread. How much copper was in the steel-al-

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loy walls of reactor vessels across the country? Reactor-vessel manufacturers and utilities began leafing through old files to find what information they had about the copper content of metals in reactors.

Records showed that there was some copper in the vessel walls themselves. "We used a lot of auto stock," explained Marston. "When you melt it, you can't get all the wiring out."

But welds in vessel walls were the real problem. Before the industry realized what was happening, which was about 1972, spools of copper-coated welding wire were routinely used for these welds. "The copper was used to prevent rust," noted Stephen H. Hanauer, director of safety technology at the NRC. "Someone probably got a $10 prize for the suggestion."

Reactor builders switched to nickel-coated electrodes, but they couldn't replace the welds in older reactors. When I visited Marston last winter, the significance of those welds became clear. On his desk was a slab of metal that looked like a paperweight gone wild. I thought it was eight inches wide. But it was really eight inches thick—the thickness of a reactor-vessel wall. The weld was a yellowish stripe in the steel, tapering from three inches thick on one side to two inches on the other. Marston told me that it can take three weeks of repeated passes with electrodes to complete one of those welds. That type of weld, engineered to be a powerful bond between huge steel sections of reactor vessels, contained enough copper to become a potential hazard instead.

Interest in reactor-vessel embrittlement heated up in 1977, Marston recalls. There was trouble with the sample holders in a reactor built by Babcock and Wilcox, one of the major suppliers, he says. Vibration kept knocking them loose. All the samples were taken out, and "it looked worse than we thought," Marston said, indicating that embrittlement was progressing faster than expected in the test samples.

Added to this continued confirmation of embrittled-metal samples and copper contamination of vessels was an event the following year that, for some, increased the alarm.

On March 20, 1978, a worker at the Rancho Seco nuclear generating plant near Sacramento, Calif., dropped a light bulb into an instrument panel. The panel shorted out and the plant's instruments went haywire, flashing fake signals to the control systems. Rancho Seco's emergency cooling system kicked into operation. Cold water

Continued

flooded into the reactor, dropping the temperature from 582 degrees F to 285 in a little more than an hour.

Pressure inside the reactor vessel first dropped from the normal 2,200 pounds per square inch to under 1,600 psi. Then, as high-pressure water pumps were triggered, the pressure went back over 2,000 psi. With no reliable instrumentation to guide them, control-room technicians kept the cold water flowing, maintaining the combination of unexpectedly low temperature and high pressure for several hours.

The Rancho Seco "transient," as nuclear engineers call it, made it clear that pressurized-water reactors were susceptible to abrupt changes in temperature and pressure. Could any pressurized reactors already have small cracks? And could vessel walls containing such cracks, subjected to sudden changes of temperature and pressure during an accident, then rupture, draining the coolant water and producing a catastrophic meltdown of the core?

The truth is that nobody knows for certain. Calculations indicate that under pressurized-thermal-shock conditions, a reactor vessel will fail only if cracks of a certain dimension are present on the inside wall. Inspections throughout the industry have used ultrasound and other nondestructive testing methods and thus far have found no such cracks. Industry representatives say they are reasonably confident that no cracks are there. Critics say the inspection equipment isn't good enough to detect the cracks. The NRC says its analyses assume that some cracks exist, no matter what inspections show.

Richard Cheverton of the Oak Ridge National Laboratory, whose team has performed many of the thermal-shock analyses, says assumptions about weaknesses in nuclear power plants had to be made. Take the critical issue of cracks in the reactor-vessel walls. "It's difficult to look for flaws after the reactor is in operation, and it's still a question of how good a job one can do," Cheverton said. "It's not clear yet whether some of the shallow flaws that can get us into trouble can be found with accuracy, so we tend to assume that the flaws will be there."

But Richard J. Sero, who heads a program on thermal shock for Westinghouse (a major plant builder), maintains that there is growing evidence to support the belief that the cracks aren't there. Engineers often inspect working-reactor vessels with ultrasound equipment, whose echoes are analyzed to detect anything unusual in the vessel wall—a crack, an inclusion of different material in the metal, an unevenness in the surface.

Ultrasound inspection is complicated somewhat by the fact that reactor vessels have a ½-inch-thick cladding—a permanently bonded layer—of stainless steel on the inside surface that can produce false echo patterns. But that's not an insuperable problem. Sero says he's impressed by the sensitivity of the equipment.

"We've done about a half-dozen full-vessel inspections," Sero said. "You do pick up what we call 'indications'—as many as 20 in some vessels. When you pick up any anomalies at all, you must look at your pre-service inspection to see if they existed before and what size they were.

"We've found that the equipment can pick up things like layers in the cladding," Sero continued. "When we've gone to the inspection reports, we've found that there are layers in the cladding at the same depth of the indication. Our conclusion is that in all the inspections we've done, we haven't found any indications that we can't resolve as inclusions of different material or layers."

> "The NRC may consult its Ouija board and get a number, but the error bands are so large, it's useless"

Sero says Westinghouse gained confidence in the inspection results when one test showed a gouge on the outside wall of a reactor vessel. "We were able to get pictures of the reactor vessel that were taken before it was installed," he said. "We found that it was a gouge that existed before it went to the plant." A sample of a vessel wall containing a crack is used to calibrate instruments.

The NRC recently released a detailed study on pressurized thermal shock and reactor safety. If you really want a good fight, ask people about the reliability of those safety estimates. The method the NRC and the industry uses is called probabilistic risk assessment. It's designed to get around a rather impressive lack of concrete evidence. All the calculations about pressurized thermal shock, for example, are based on just eight events that have occurred at nuclear plants, including the Rancho Seco transient and the most famous incident of all, Three Mile Island.

In a probabilistic risk assessment, you estimate the likelihood of an event that initiates a transient, then estimate the likelihood of the reaction to that event, the reaction to that reaction, and so on down the line.

Westinghouse, *for example*, has a computer analysis that starts with 17 possible initiators and runs through event trees to more than 6,200 end points. The NRC has done the same thing. Its numbers come out more or less in agreement about the risk of thermal shock. But there are inevitable differences of opinion about the value of those calculations, which show that although there is no clear and present danger, corrective action should be taken at some reactors to reduce the hazard of thermal shock.

Not everyone agrees with the calculations. "The NRC may consult its Ouija board and come up with a number," said Robert Pollard of the Union of Concerned Scientists, "but the error bands on it are so large that it's essentially useless."

That's not exactly so, says Cheverton of Oak Ridge. "It's possible to estimate what the uncertainty in the analysis is, and you have to live with that uncertainty," he said. "But you take the conservative end of it and work with that."

A lack of data is more or less conceded all through the NRC report. "Perhaps the most significant uncertainty in the treatment . . . is that there are known low-frequency potential over-cooling events much more severe than those that have occurred," the report says at one point. "Because these events have not occurred, they have not been taken into account in the frequency distribution." In other words, it's tough to predict the possibility of something that has never happened. In another section, the report notes "substantial uncertainties" in some estimates and calculations that are uncertain by "plus or minus at least two orders of magnitude, a broad band of uncertainty, indeed."

What else can we do? the NRC people ask. "It isn't well defined, but it's the best information we have," said the NRC's Hanauer.

Your best is none too good, the critics say. They point out that the probabilistic-risk-assessment technique is the same one used in the famous Rasmussen report of 1974, in which a team headed by MIT professor Norman Rasmussen calculated the risks of nuclear accidents. Rasmussen came up with some comfortingly low-risk figures. Just last year, though, the

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NRC looked over the operating data that have accumulated since then and concluded that the odds of a nuclear accident occurring calculated by Rasmussen were low by a factor of 30.

Hanauer says that risk calculators have learned a lot from Rasmussen's pioneering effort. "He kissed off earthquakes in two pages and floods in two lines," Hanauer noted. Taking one volume of a shelf-long safety assessment of the Indian Point reactor near New York City, Hanauer pointed out that earthquakes and floods were toward the top of the list of risks. The NRC has learned to include such risks in its risk assessments, Hanauer says.

But Basdekas dismisses the report as "the quantification of wishful thinking." And George Sih, director of the Institute of Fracture and Solid Mechanics at Lehigh University, says that the impressive report is built on a foundation of sand.

"The samples they study are five inches long, and the vessels are 600 inches long," Sih said. "The sample is very thin, and the vessel is eight inches thick. We don't know how to transfer small-sample data to the design of large-scale structural components. The scaling effect in size and also the scaling effect in time are among the most difficult questions we have."

If critics think the NRC has been too speculative, industry believes the report is too conservative. You can arrive at just about any conclusion you want by putting in the appropriate numbers, Marston says. "By changing the assumptions," he explained, "I can show that one of these things has no useful life at all or a lifetime of 30 to 40 years." The NRC consistently takes the most conservative numbers for its estimates, he says.

One of the key factors that the NRC's experts looked at was the transition temperature at which a piece of metal stops being ductile and becomes brittle enough to break easily. A crucial part of the NRC report was to set a point at which this transition temperature in a given reactor would be cause for concern. The report sets the danger point at 300 degrees F for vertical welds, 270 degrees for horizontal ones.

Higher transition temperatures are worse, since the reactor vessel must be maintained at these temperatures if the effects of brittle metal are to be avoided. The original standard for nuclear reactors was no more than 200 degrees F. The temperature is higher for vertical welds because pressure tends to force the welds out, increasing the possibility that a crack will break through the vessel wall.

Determining a transition temperature depends on the composition of a metal, the amount of radiation it receives, and, most controversially, the stresses to which it is exposed. The NRC staff used a formula to predict how assumed pre-existing cracks might extend into the vessel wall.

As a result of tests on the rate of embrittlement at various plants, the NRC predicted when some of them will reach a danger point. All things considered, the NRC report reached a reasonably comforting conclusion. It listed 40 pressurized-water reactors in which pressurized thermal shock was an issue. "If no one does anything, we've got one reactor that's in big trouble, four others that are a little behind it, and four that are in a mild kind of trouble," Hanauer told me. "The rest of them will not reach

> **"Though the inner portion is brittle, the outer portion is tough; radiation damage in the wall is attenuated"**

the screening criterion [the transition temperature] during the anticipated life of the plant."

The "big-trouble" generating plant is the H. B. Robinson 2 reactor of Carolina Power and Light. Hanauer calculated that if nothing were done, it would reach the transition-temperature criterion in September of 1987. Turkey Point 3 and 4 in Florida get there in 1988; Calvert Cliffs 1 in Maryland gets there in 1989; and Fort Calhoun in Nebraska arrives in 1990. Rancho Seco, Maine Yankee, Oconee 2 in South Carolina, and Three Mile Island 1 arrive in the 1990s. Everything else is 21st century, Hanauer says.

Reactor manufacturers accepted those numbers without too much argument. "Their conclusions are more or less in line with ours," said Sero of Westinghouse. Sero says that Westinghouse thinks the NRC could set its transition-temperature numbers about 30 degrees lower, but he isn't arguing with the basic premises of the report.

Nuclear critics are. They center their fire on the vast number of assumptions that had to be made in the report because information about the probability of different events occurring and about the reliability of safety systems simply isn't available. Rep. Markey's reaction, for example, was that the risk-assessment technique was "like predicting the winner of the World Series after the first exhibition game."

There's also a lot that the utilities and manufacturers can do to lessen any possible danger, industry experts say. One easy step is to reshuffle the fuel elements in the reactor core, putting older fuel elements, which emit fewer neutrons, close to the vessel wall. "It's easy and cheap to reduce neutron flux by a factor of two," acknowledged Hanauer.

Critics say that repositioning the fuel elements isn't enough. They want American utilities to reduce neutron exposure even further by inserting dummy fuel elements next to the vessel wall. That's been done at two reactors in West Germany and one Russian-built reactor in Finland. But utilities are reluctant to take the reduction in generating capacity that dummy fuel elements bring.

There are many other steps that can be taken, Marston said. One is the marvelously simple measure of heating the emergency cooling water to reduce thermal shock. Keeping the emergency water supply at 120 degrees F rather than room temperature is cheap and effective, Marston says. Thermal shock can also be reduced by adding controls to throttle back the automatic-feedwater system, he notes.

Improved training for reactor operators is another industry option. The idea is to get them ready for all the problems that could lead to a significant transient, then avoid the sequences that end in serious trouble.

The last resort is annealing. The reactor would be shut down, all the fuel elements would be removed, and the vessel would be heated to 850 degrees F for a week. A study done by Westinghouse for the Electric Power Research Institute concluded that annealing would make the vessel walls young again. The process isn't cheap. One report cited costs of $60 million or more for a single reactor, including the price of the electricity that the plant did not generate during the treatment.

No one is thinking about annealing right now. Instead, utilities and manufacturers are making detailed studies of all the factors affecting the thermal-shock issue for individual plants. The NRC report has asked for such a plant-specific report at least three years before a reactor reaches its screening criterion for danger.

For the Robinson 2 reactor, the report would be due in 1984. Carolina Power and Light is hard at work, says

Thomas S. Elleman, who is in charge of nuclear safety. The vessel wall has been inspected, and no cracks were found. New training for reactor personnel is under way. The company is studying a proposal to heat the emergency water supply.

Neutron exposure has been reduced by putting the older fuel elements next to the reactor wall. How much extra time will the program buy? "It's premature to speculate about that," Elleman said.

There's no panic at the NRC, the manufacturers, or the utilities. The problem is well understood, Cheverton says, and the Oak Ridge analysis indicated that even if worse came to worst, a reactor vessel would not break wide open. "Even though the inner portion is brittle, the outer portion still is relatively tough because the radiation damage is attenuated through the wall," Cheverton said. "A crack might be driven through the inner part, but it tends to arrest at the outer part."

But that assessment could easily be wrong, says Pollard of the Union of Concerned Scientists. "There's no dispute that current emergency systems would not be able to cope with a fracture of the reactor vessel," he said. "For other problems, you can make a reasonable argument that you have some defense in depth. The defense-in-depth philosophy disappears when you talk about pressurized thermal shock."

The real problem, Pollard says, is that the nation's nuclear regulators and the manufacturers allowed a major construction program to roar ahead without considering the range of unknown dangers that lay before them.

"The Atomic Energy Commission went forward with all this undue optimism," complained Pollard, who resigned from his job as a regulator years ago in disgust. "Now we're in a position where nothing can be done to correct the mistakes without causing someone undue harm. I expected them to do the job back in the 1960s. Now everyone but the nuclear industry has to suffer."

"My perception is that the problem is well in hand," said Westinghouse's Sero. "We have significant research programs under way, we are putting significant money and engineering efforts into it, and we have a firm understanding that is going to improve, which will show that our predictions were very conservative." ■

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 27, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON PALISADES PLANT

Dear Dr. Seaborg:

At a Special Meeting, January 23-24, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consumers Power Company for authorization to operate the Palisades Plant at power levels up to 2200 MWt. This project was also considered at the 113th ACRS meeting, September 4-6, 1969, the 115th ACRS meeting, November 6-8, 1969, and the 116th ACRS meeting, December 11-13, 1969. Subcommittee meetings were held on July 31, 1969, at the site, and on October 29, 1969, December 3, 1969, and January 22, 1970, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of Consumers Power Company, Combustion Engineering, Inc., Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on the construction of this plant in its letter dated January 18, 1967.

The site for the Palisades Plant consists of 487 acres on the eastern shore of Lake Michigan in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The minimum exclusion radius for the site is 2300 feet and the nearest population center of more than 25,000 residents consists of the cities of Benton Harbor and St. Joseph, Michigan, which are approximately 16 miles south of the site.

The nuclear steam supply system for the Palisades Plant is the first of the Combustion Engineering line currently licensed for construction. A feature of the Palisades reactor is the omission of the thermal shield. Studies were made by the applicant to show that omission of the shield would not adversely affect the flow characteristics within the reactor vessel or alter the thermal stresses in the walls of the vessel in a manner detrimental to safe operation of the plant. Surveillance specimens in the vessel will be used to monitor the radiation damage during the life of the plant. If these specimens reveal changes that affect the safety of the plant, the reactor vessel will be annealed to reduce

radiation damage effects. The results of annealing will be confirmed by tests on additional surveillance specimens provided for this purpose. Prior to accumulation of a peak fluence of 10^{19} nvt (> 1 Mev) on the reactor vessel wall, the Regulatory Staff should reevaluate the continued suitability of the currently proposed startup, cooldown, and operating conditions.

The secondary containment is a reinforced concrete structure consisting of a cylindrical portion prestressed in both the vertical and circumferential directions, a dome roof prestressed in three directions, and a flat non-prestressed base. Before operation, it will be pressurized and extensive measurements will be made of gross deformations and of strains in the linear, reinforcement, and concrete, and the pattern and size of cracks in the concrete will be observed and measured. The applicant has proposed suitable acceptance criteria for the pressure test, and the ACRS recommends that the Regulatory Staff review and assess the results of this test prior to operation at significant power.

The prestressing tendons in the containment consist of ninety, one-quarter-inch diameter wires. They are not grouted or bonded, and are protected from corrosion by grease pumped into the tendon sheaths. The applicant has proposed that selected tendons be inspected periodically for broken wires, loss of prestress, and corrosion. If degradation is detected, the inspection can be extended to the remaining tendons, all of which are accessible. The applicant is performing studies to determine the appropriate number and interval for tendon inspection. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The core is calculated to have a slightly negative moderator coefficient at full power operation at beginning-of-life, but uncertainties in the calculations are such that the existence of a positive moderator coefficient cannot be precluded. The applicant has stated that the moderator coefficient will not exceed $+0.5 \times 10^{-4}\ \Delta k/k/^{o}F$ at beginning-of-life, computed from start-up test data on a conservative basis. The applicant also plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses required to establish the value of the moderator coefficient.

The meteorological observation program conducted at the site subsequent to the Committee's report to you on January 18, 1967, indicated the need for the addition of iodine removal equipment to the containment for use in the unlikely event of a loss-of-coolant accident. The applicant proposed to install means for adding sodium hydroxide to the water in the containment spray system. However, because of uncertainties regarding the generation of hydrogen and the effects of other materials resulting

from the reaction of this alkaline solution with the relatively large amounts of aluminum in the containment, this spray additive will not be used unless it can be shown by further studies that the use of sodium hydroxide is clearly acceptable. In addition, the applicant will carry out studies of iodine removal by borated water sprays without sodium hydroxide. If the results of these studies are not acceptable, a different iodine removal system satisfactory to the Regulatory Staff will be installed at the first refueling outage. A report on the applicant's plans will be submitted to the AEC within six months following issuance of a provisional operation license. The Committee believes that this procedure is satisfactory for operation at power levels not exceeding 2200 MWt.

The applicant has stated that if fewer than four primary coolant pumps are operating, the reactor overpower trip settings will be reduced such that the safety of the reactor is assured in the absence of automatic changes in the thermal margin trip settings.

The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. Studies are to be made on further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant should consider the results of such studies and incorporate appropriate provisions in the Palisades Plant.

The Committee recommends that attention be given to the long-term ability of vital components, such as electrical equipment and cables, to withstand the environment of the containment in the unlikely event of a loss-of-coolant accident. This matter is applicable to all large, water-cooled power reactors.

Continuing research and engineering studies are expected to lead to enhancement of the safety of water-cooled reactors in other areas than those mentioned: for example, by determination of the extent of the generation of hydrogen by radiolysis and from other sources, and development of means to control the concentration of hydrogen in the containment, in the unlikely event of a loss-of-coolant accident; by development of instrumentation for inservice monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system; and by evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to these problems develop and are evaluated

by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Palisades Plant can be operated at power levels up to 2200 MWt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References:

1. Final Safety Analysis Report for the Palisades Plant

2. Amendments No. 9-19 to license application

LBP-15-17

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Ronald M. Spritzer, Chair
Dr. Gary S. Arnold
Dr. Thomas J. Hirons

In the Matter of:	Docket No. 50-255-LA
ENTERGY NUCLEAR OPERATIONS, INC.	ASLBP No. 15-936-03-LA-BD01
(Palisades Nuclear Plant)	May 8, 2015

MEMORANDUM and ORDER
(Ruling on Petition to Intervene and Request for a Hearing)

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NUCLEAR REGULATORY COMMISSION

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MEMORANDUM and ORDER
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I. Introduction

Before the Licensing Board is a petition to intervene and request for a hearing filed by Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future – Shoreline Chapter (Shoreline), and the Nuclear Energy Information Service (NEIS) (collectively Petitioners).[1] We find that Petitioners have established representational standing to intervene in this proceeding. We do not, however, admit Petitioners' contention. Because Petitioners have not proffered an

[1] Petition to Intervene and for a Public Adjudication Hearing of Entergy License Amendment Request for Authorization to Implement 10 CFR §50.61a, 'Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events' (Dec. 1, 2014). Petitioners amended their petition on December 8, 2014, and indicated that the sole difference in the amended petition "is correction of the initial Federal Register reference as it appeared on page 1 of the December 1 filing to reflect Vol. 79 instead of Vol. 78." Amended Petition to Intervene and for a Public Adjudication Hearing of Entergy License Amendment Request for Authorization to Implement 10 CFR §50.61a, 'Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events' at 1 n.1 (Dec. 8, 2014) [hereinafter Amended Petition]. The Board references the Amended Petition throughout this Memorandum and Order.

admissible contention, they have not satisfied the prerequisites for the Board to grant their hearing request.[2]

II. Procedural Background

This proceeding concerns Entergy Nuclear Operations, Inc.'s (Entergy's) request to amend the operating license for the Palisades nuclear plant (Palisades).[3] Palisades is a single-pressurized water reactor (PWR) facility located on the eastern shore of Lake Michigan, five miles south of South Haven, Michigan.[4] The requested amendment would permit Entergy to use an alternate method to evaluate the minimum fracture toughness required by the Palisades reactor pressure vessel (RPV) to safely withstand a pressurized thermal shock (PTS) event.[5] That alternate method is set forth in an agency regulation, "Alternate fracture toughness requirements for protection against pressurized thermal shock events."[6]

In an operating nuclear power plant, the reactor vessel is continuously exposed to neutrons from fission reactions occurring inside the vessel.[7] Over time, this neutron radiation

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[2] See 10 C.F.R. § 2.309(a), (f)(1).

[3] License Amendment Request to Implement 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (July 29, 2014) (ADAMS Accession No. ML14211A524) [hereinafter LAR].

[4] NRC Staff Answer to Petition to Intervene and Request for a Hearing Filed By Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future–Shoreline Chapter, and the Nuclear Energy Information Service at 2 (Jan. 12, 2015) [hereinafter NRC Staff Answer].

[5] See LAR, attach. 1 at 1. Entergy enclosed within its LAR a technical report designed "to provide Palisades with the basis for implementation of the" amended PTS screening program. See Westinghouse, Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades at v (June 2014) (ADAMS Accession No. ML14211A525) [hereinafter Palisades Alternate PTS Rule Evaluation].

[6] 10 C.F.R. § 50.61a [hereinafter the Alternate PTS Rule] (emphasis removed); see also Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Final Rule, 75 Fed. Reg. 13, 14 (Jan. 4, 2010).

[7] Division of Fuel, Engineering and Radiological Research, Office of Nuclear Regulatory Research, Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61) Summary Report, NUREG-1806 at xix (Aug. 2007),

embrittles the RPV walls, making them less able to resist fracturing, i.e., "fracture toughness" decreases.[8] If there is a flaw in a reactor vessel wall that is embrittled due to neutron exposure, certain events can cause the flaw to propagate through the wall, resulting in a breach of the RPV and a possible accident.[9] Of significant concern is a PTS event, which is "characterized by a rapid cooling (i.e., thermal shock) of the internal RPV surface and downcomer, which may be followed by repressurization of the RPV."[10] The possible triggers of a PTS event include "a pipe break or stuck-open valve in the primary pressure circuit," or "a break of the main steam line."[11]

On September 30, 2014, the NRC Staff (the Staff) published notice of Entergy's LAR,[12] and concluded that the LAR presents "no significant hazards consideration" under 10 C.F.R. § 50.92(c).[13] In response to the LAR notice, Petitioners filed the instant petition to intervene and request for a hearing.[14]

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available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1806/v1/ [hereinafter Alternate PTS Rule Technical Basis Report].

[8] Id. at xx.

[9] See id. at xix.

[10] Id.

[11] Id.; see also 75 Fed. Reg. at 14. As the Alternate PTS Rule Technical Basis Report further explains, during these scenarios, "the water level in the core drops as a result of" depressurization or leaks. Alternate PTS Rule Technical Basis Report at xix. Emergency makeup water is then added to the reactor cooling loop, either manually or automatically, to keep the reactor core covered with water. Id. As the makeup water is much colder than the water in the reactor, a rapid cooling of the outside reactor wall results. Id. For over-embrittled RPVs, the temperature shock "could be sufficient to initiate a running crack, which could propagate all the way through the vessel wall." Id. As the reactor is still producing heat, even in a shutdown mode, the RPV could re-pressurize, adding additional stress to the already-propagating crack. See id. at xix, xxiv, xxv ("A major contributor to the risk-significance of [certain PTS events] is the return to full system pressure" after cold makeup water is introduced. This could occur, for example, when a stuck-open valve recloses.).

[12] Biweekly Notice, Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations, 79 Fed. Reg. 58,812, 58,814–16 (Sept. 30, 2014).

[13] Id. at 58,815 ("The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff

Petitioners' statement of their contention is:

> The licensing framework that the NRC is applying to allow Palisades to continue to operate until August 2017 includes both non-conservative analytical changes and mathematically dubious comparisons to allegedly similar "sister" reactor vessels. Palisades' neutron embrittlement dilemma continues to worsen as the plant ages, and Palisades has repeatedly requested life extensions which have ignored and deferred worsening embrittlement characteristics of the RPV for decades. Presently, Entergy plans to deviate from the regulatory requirements of 10 C.F.R. § 50.61 to §50.61a (Alternate Fracture Toughness Requirements). This new amendment request introduces further non-conservative analytical assumptions into the troubled forty-three (43) year operational history of Palisades. Entergy's License Amendment Request (LAR) contains an equivalent margins evaluation, which is an untried methodological approach.[15]

Petitioners' hearing request was referred to this Board for consideration.[16] Both Entergy and the Staff have filed answers opposing the Amended Petition,[17] to which Petitioners have filed a reply.[18] On March 25, 2015, the Board heard oral argument on standing and contention admissibility.[19]

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proposes to determine that the amendment request involves no significant hazards consideration.").

[14] Amended Petition.

[15] Id. at 11–12.

[16] Entergy Nuclear Operations, Inc.; Establishment of Atomic Safety and Licensing Board, 79 Fed. Reg. 77,041 (Dec. 23, 2014); see also Memorandum from Richard J. Laufer, Acting Secretary of the Commission, to E. Roy Hawkens, Chief Administrative Judge, Atomic Safety & Licensing Board Panel, Referring the Amended Petition to the Atomic Safety & Licensing Board Panel for Disposition (Dec. 11, 2014).

[17] Entergy's Answer Opposing Petition to Intervene and Request for Hearing (Jan. 12, 2015) [hereinafter Entergy Answer]; NRC Staff Answer.

[18] Petitioners' Combined Reply in Support of Amended Petition to Intervene and for a Public Adjudication Hearing of Entergy License Amendment Request for Authorization to Implement 10 CFR §50.61a, 'Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events' (Jan. 20, 2015) [hereinafter Reply].

[19] Transcript of Oral Argument on Contention Admissibility (Mar. 25, 2015) [hereinafter Tr.].

III. Regulatory Background

A. The 1985 PTS Rule & Embrittlement Screening Program (10 C.F.R. § 50.61)

In 1985, the NRC implemented a mandatory program to monitor PWR RPVs for embrittlement over time, coupled with screening limits to prevent over-embrittled reactors from operating.[20] The program to monitor PWR RPVs is described in 10 C.F.R. Part 50, Appendix H, and is titled "Reactor Vessel Material Surveillance Program Requirements" (Surveillance Program).[21] The purpose of the Surveillance Program "is to monitor changes in the fracture toughness properties of ferritic materials [iron-based metals, such as steel] . . . which result from exposure of these materials to neutron irradiation and the thermal environment."[22] The Surveillance Program relies on physical material samples, also known as specimens, capsules, or coupons,[23] "which are withdrawn periodically from the reactor vessel."[24] The NRC must pre-approve the schedule for removing material samples from the reactor vessel.[25]

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[20] See Analysis of Potential Pressurized Thermal Shock Events, Final Rule, 50 Fed. Reg. 29,937 (July 23, 1985) (creating the screening criteria); Fracture Toughness and Surveillance Program Requirements, Final Rule, 38 Fed. Reg. 19,012 (July 17, 1973) (creating the program to monitor PWR RPVs).

[21] 10 C.F.R. pt. 50, app. H (capitalization modified).

[22] Id. pt. 50, app. H(I).

[23] Amended Petition at 11; Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Unit 1), CLI-93-21, 38 NRC 87, 89 (1993).

[24] 10 C.F.R. pt. 50, app. H(I). The NRC's regulations further require that the physical specimens "be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface." Id. pt. 50, app. H(III)(B)(2).

[25] Id. pt. 50, app. H(III)(B)(3). The NRC's regulations also allow for an "integrated" Surveillance Program among similar reactors, if the reactors "have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage." Id. pt. 50, app. H(III)(C). The regulations also allow for an exemption from the Surveillance Program if a reactor's lifetime irradiation levels are below a certain threshold. Id. pt. 50, app. H(III)(A) (applying to reactors which can conservatively demonstrate by experiments on similar vessels that "the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} [neutrons per centimeter squared] (E >1MeV [mega electron volt])").

While Appendix H establishes the Surveillance Program by which the RPVs are monitored for fracture toughness, the actual screening limits are established in 10 C.F.R. § 50.61, entitled "Fracture toughness requirements for protection against pressurized thermal shock events."[26] Section 50.61 establishes an analytical approach that relies on data gathered from the Surveillance Program to calculate the RPV wall's fracture toughness, and compares it with a safety limit that cannot be exceeded.[27]

In the NRC's regulations, steel fracture toughness is represented by proxy as a temperature value, known as "reference temperature." As explained by the Staff, "[r]eference temperature is the metric that the NRC uses to quantitatively assess brittleness, so these terms may be regarded as synonymous. Steel having a high 'reference temperature' also has a higher degree of brittleness than steel with a low reference temperature."[28] This is because the ability of steel to resist fracture changes as a function of temperature. When steel is at high temperatures, it can retain its ductility and related ability to resist fracturing from PTS events, even after extended periods of neutron irradiation.[29] On the other hand, at very low temperatures, steel is naturally brittle, and even unirradiated steel can potentially suffer brittle

[26] 10 C.F.R. § 50.61 (emphasis removed).

[27] See id. § 50.61(c)(2)(i) ("Results from the plant-specific surveillance program must be integrated into the [fracture toughness] estimate if the plant-specific surveillance data has been deemed credible . . . ."); Alternate PTS Rule Technical Basis Report at xx ("The surveillance results are then used together with the formulae and tables in 10 CFR 50.61 to estimate the fracture toughness" of the RPV wall.).

[28] John B. Giessner, Division of Reactor Projects, Summary of the March 19, 2013, Public Meeting Webinar Regarding Palisades Nuclear Plant, encl. 2 at 4 (Apr. 18, 2013) (ADAMS Accession No. ML13108A336) [hereinafter Palisades Webinar].

[29] See Alternate PTS Rule Technical Basis Report at xxxviii–xxxix (noting that with steel at high temperatures "cleavage cannot occur"). A "Cleavage fracture" is the type of fracture associated with fracture of brittle materials. See id. at xxxviii. The Board at times cites to certain Staff guidance documents, such as the Alternate PTS Rule Technical Basis Report, to help explain the background science behind the phenomena at issue in this proceeding. This does not mean, however, that the Board necessarily adopts the Staff's conclusions put forward in these documents as to whether Palisades' LAR meets the relevant regulatory requirements. See infra Section V(B) (Scope of Review of License Amendments).

failure.[30]

The point at which steel transitions from the high-temperature, fracture-resistant-state, to the low-temperature, brittle state, is called the "RT_{NDT}," or "Transition fracture toughness reference temperature," or more simply "reference temperature."[31] As described by Staff guidance documents, this transition point depends primarily on two factors: (i) material composition and (ii) cumulative irradiation by high-energy neutrons.[32] As steel is exposed to more high-energy neutrons (i.e., its fluence increases),[33] RT_{NDT} increases concurrently.[34] Thus, as fluence increases, the steel stays brittle at higher and higher temperatures, and it is therefore more likely to fracture as a result of PTS events.

The NRC established screening limits in 10 C.F.R. § 50.61 (the Current Screening Criteria) to reduce the risk that a PTS event will result in an RPV fracture. The screening limits are expressed as temperature values. When the reference temperature of an RPV is above this screening limit, the RPV is considered to have an unreasonably high risk of fracture from a PTS

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30 See id. at xxxviii–xxxix (noting that with steel at low temperatures, "fracture occurs by cleavage").

31 Id. at xxxiv. "NDT" stands for Nil-Ductility Temperature. Id. at xxxi.

32 Id. at xx ("[T]ransition temperatures increase as a result of irradiation damage throughout the operational life of the vessel."); id. § 2.1.3 (discussing the factors affecting fracture toughness); id. § 2.4.2 (limiting the fluence to only high-energy "fast" neutrons, which have energies above one mega electron volt).

33 Fluence is the integral of the neutron flux over time. The neutron flux is the total distance traversed by neutrons within a unit volume of material within one unit of time. Typically the unit volume is one cubic centimeter and the unit time is one second. Thus the unit of neutron flux is neutron-centimeter/centimeter^3-second, typically expressed as neutrons/centimeter^2-second. See Samuel Glasstone and Alexander Sesonske, Nuclear Reactor Engineering § 2.118 (Van Nostrand Reinhold Co. 1967).

34 See Alternate PTS Rule Technical Basis Report § 2.4.1 (discussing the index temperature approach to characterizing fracture toughness in ferritic materials).

event.[35] The PTS "screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials."[36]

If the RT_{NDT} values projected at specific areas of the RPV for the end of life of the plant, known as RT_{PTS},[37] surpass the Current Screening Criteria, the licensee must submit a safety analysis and obtain the approval of the Office of Nuclear Reactor Regulation to continue to operate.[38] If that office does not approve continued operation based on the licensee's safety analysis, the licensee must request an opportunity to modify the RPV or related reactor systems to "reduce the potential for failure of the reactor vessel due to PTS events."[39]

B. The Alternate PTS Rule & Embrittlement Screening Program (10 C.F.R. § 50.61a)

While no reactor is expected to exceed the Current Screening Criteria established in Section 50.61 during its 40 year operating license, some plants "are likely to exceed the screening criteria during the extended period of operation of their first license renewal."[40] The Staff has noted that Palisades in particular is one of the first plants likely to exceed the Current Screening Criteria, as Palisades' RPV is "constructed from some of the most irradiation-

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[35] See 10 C.F.R. § 50.61(b)(2). The Current Screening Criteria "correspond to a limit of 5 x 10^{-6} events/year on the annual probability of developing a through-wall crack" in the RPV. Alternate PTS Rule Technical Basis Report at xx.

[36] 10 C.F.R. § 50.61(b)(2); see also 75 Fed. Reg. at 13 ("The current PTS rule . . . establishes screening criteria below which the potential for a reactor vessel to fail due to a PTS event is deemed to be acceptably low.").

[37] 10 C.F.R. § 50.61(a)(7) ("RT_{PTS} means the reference temperature, RT_{NDT}, evaluated for the [end of life] Fluence for each of the vessel beltline materials."); Alternate PTS Rule Technical Basis Report § 11.2 ("10 CFR 50.61 defines RT_{PTS} as the maximum RT_{NDT} of any region in the vessel (a region is an axial weld, a circumferential weld, a plate, or a forging) evaluated at the peak fluence occurring in that region.").

[38] 10 C.F.R. § 50.61(b)(3)–(5).

[39] Id. § 50.61(b)(6).

[40] 75 Fed. Reg. at 13.

sensitive materials in commercial reactor service today."[41] This concern, as well as significant advancements in failure analysis and materials knowledge, prompted the NRC to reexamine the Section 50.61 approach for projecting fracture toughness and the Current Screening Criteria.[42]

In August 2007, the NRC issued NUREG-1806, "Technical Basis for Revision of the [PTS] Screening Limit in the PTS Rule (10 CFR 50.61)." That report summarized the results of a five year study by the NRC, the purpose of which "was, to develop the technical basis for revision of the Pressurized Thermal-Shock (PTS) Rule."[43] The report concluded that through-wall cracks were much harder to create in RPVs than initially thought, and occurred in fewer circumstances.[44] The report thus recommended a more detailed approach to setting screening criteria that would take into account the varying conditions along different parts of the RPV.[45] The report also recommended removing the "margin term" that had been included in the Current Screening Criteria to account for unknown factors, because essentially all factors are now known and are effectively quantified.[46]

On October 3, 2007, the Staff published a notice of proposed rulemaking.[47] The rulemaking notice stated that the Alternate PTS Rule Technical Basis Report "conclude[d] that the risk of through-wall cracking due to a PTS event is much lower than previously estimated,"

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[41] Alternate PTS Rule Technical Basis Report at xxii.

[42] See Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Proposed Rule, 72 Fed. Reg. 56,275, 56,276 (Oct. 3, 2007); Alternate PTS Rule Technical Basis Report at iii, xx–xxiii.

[43] Alternate PTS Rule Technical Basis Report at xix.

[44] See id. at xx–xxiii.

[45] Id. at xxv ("Specifically, we recommend a reference temperature for flaws occurring along axial weld fusion lines (RT_{AW} or $RT_{AW\text{-}MAX}$), another for flaws occurring in plates or in forgings (RT_{PL} or $TR_{PL\text{-}MAX}$), and a third for flaws occurring along circumferential weld fusion lines (RT_{CW} or $RT_{CW\text{-}MAX}$).").

[46] Id. at xxvii.

[47] 72 Fed. Reg. 56,275.

and that "[t]his finding indicates that the screening criteria in 10 CFR 50.61 are unnecessarily conservative."[48] On January 4, 2010, the NRC issued the final rule, creating 10 C.F.R. § 50.61a.

The Alternate PTS Rule makes two important changes.[49] First, Section 50.61a replaces the relatively broad Current Screening Criteria (270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials) with more detailed Alternate Screening Criteria.[50] The Alternate Screening Criteria consist of eighteen different reference temperature limits that depend on RPV wall thickness and the part of the RPV under consideration.[51]

The Alternate PTS Rule also changes how licensees derive projected reference temperatures for the components of their RPVs.[52] Section 50.61a relies on a probabilistic "embrittlement model" to predict future reference temperatures across the RPV, which is then verified by existing surveillance data in a process called the "consistency check."[53] Section 50.61, by contrast, continuously integrates surveillance data into future embrittlement projections.[54]

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[48] Id. at 56,276.

[49] Otherwise, like the old rule, the new rule provides measures for ongoing reporting, 10 C.F.R. § 50.61a(d)(1), and mitigation processes for licensees if they project they will exceed (or they do exceed) the Alternate PTS Rules' screening criteria. Id. § 50.61a(d)(2)–(7).

[50] 75 Fed. Reg. at 18.

[51] 10 C.F.R. § 50.61a(g) tbl. 1.

[52] See Id. § 50.61a(f), (f)(6)(B)(ii).

[53] Id.

[54] Compare id. § 50.61a(f)(6)(i) (requiring that a licensee perform a "consistency check" of its embrittlement model against available surveillance data), and Alternate PTS Rule Technical Basis Report § 3.1.1 (The Alternate PTS Rule is designed to "enable all commercial PWR licensees to assess the state of their RPVs relative to such a new criterion without the need to make new material property measurements," instead using "only information that is currently available."), with 10 C.F.R. § 50.61(c)(2)(i) (requiring that "plant-specific surveillance data must be integrated into the RT_{NDT} estimate"), and Alternate PTS Rule Technical Basis Report § 2.4.2

In the final rulemaking notice, the Commission concluded that the new "estimation procedures provide a better (compared to the existing regulation) method for estimating the fracture toughness of reactor vessel materials over the lifetime of the plant."[55] The final rulemaking notice stated that the Alternate PTS Rule "provides reasonable assurance that licensees operating below the screening criteria could endure a PTS event without fracture of vessel materials, thus assuring integrity of the reactor pressure vessel."[56] Furthermore, the final rulemaking stated that "[t]he final rule will not significantly increase the probability or consequences of accidents, result in changes being made in the types of any effluents that may be released off site, or result in a significant increase in occupational or public radiation exposure."[57]

C. Applying to Use the Alternate PTS Rule

To take advantage of the Alternate PTS Rule, a licensee must request approval from the Office of Nuclear Reactor Regulation, in accordance with the procedures for submitting a license amendment under 10 C.F.R. § 50.90. The application must contain: (i) under Section 50.61a(f), the projected embrittlement reference temperatures along various portions of the RPV, from now to a future point, compared to the Alternate Screening Criteria; and (ii) under Section 50.61a(e), an assessment of flaws in the RPV.[58]

In calculating embrittlement reference temperatures under Section 50.61a(f), a licensee must calculate neutron flux through the RPV "using a methodology that has been benchmarked

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(Under the Current PTS Rule, material samples "from RPV surveillance programs provide the empirical basis to establish embrittlement trend curves . . . .").

[55] 75 Fed. Reg. at 18.

[56] Id. at 22.

[57] Id.

[58] 10 C.F.R. § 50.61a(c)(1)–(2). Under Section 50.61a, the licensee must separately examine for flaws in the reactor vessel. Id. § 50.61a(c)(2). The analysis of flaws in the Palisades RPV is not in dispute in this proceeding.

to experimental measurements and with quantified uncertainties and possible biases."[59] From that point, the licensee must establish $RT_{NDT(U)}$ for various key points along the RPV.[60] Then a licensee uses a series of equations and charts provided in the rule to create an embrittlement model. That model projects the reference temperatures for various parts of the RPV at the end of life of the plant, known in the new rule as RT_{MAX-X}.[61] The embrittlement model allows for calculations of RT_{MAX-X} across the RPV using probabilistic analyses, without having to rely on measured data.[62] The RT_{MAX-X} values are compared to the Alternate Screening Criteria to determine whether the RPV is safe to operate.[63]

Importantly, as calculations of RT_{MAX-X} are made analytically, without directly incorporating surveillance data, licensees have to verify that their calculations at the time of the application match up with surveillance data.[64] To do so, licensees have to perform the "consistency check" of their calculations for specific materials against "heat-specific surveillance data that are collected as part of 10 CFR part 50, Appendix H, surveillance programs."[65] The purpose of the check is to "determine if the surveillance data show a significantly different trend

[59] Id. § 50.61a(f).

[60] Id. § 50.61a(f)(4). $RT_{NDT(U)}$ is the nil-ductility reference temperature for the RPV material in the annealed state, before the reactor was operational. Id. If measured values are not available, a licensee can use a set of generic mean values. Id. § 50.61a(f)(4)(i), (ii).

[61] Id. § 50.61a(f)(1)–(3). "RT_{MAX-X} is the equivalent term for RT_{PTS} in 10 CFR 50.61a." Proposed Rulemaking — Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (RIN 3150-AI01), SECY-07-0104 (June 25, 2007).

[62] See supra note 54.

[63] See 10 C.F.R. § 50.61a(c)(3).

[64] Id. § 50.61a(f)(6)(i).

[65] 75 Fed. Reg. at 16. The regulatory history of the Alternate PTS Rule and associated draft guidance indicates that uncertainty in surveillance data measurements may be a concern, which licensees' applications should address. See id. at 16–17 (discussing potential concerns with variability in surveillance data); Regulatory Guidance on the Alternate Pressured Thermal Shock Rule, Draft Regulatory Guide DG-1299 at 12 (Mar. 2015) [hereinafter DG-1299] ("The input variables to [the equations comprising the consistency check] are subject to variability and are often based on limited data," particularly fluence.).

than the embrittlement model predicts."[66] The check includes three statistical analyses that compare the model's inputs, fluence and material properties, with the model's output, reference temperature.[67]

The consistency check is required "[i]f three or more surveillance data points measured at three or more different neutron fluences exist for a specific material."[68] The surveillance data must consist of material samples that are the same composition, or "heat," as the materials being evaluated by the model.[69] The surveillance data, however, need not be obtained from the same RPV that is the subject of the license amendment: "Surveillance data means any data that demonstrates the embrittlement trends for the beltline materials, including, but not limited to, surveillance programs at other plants with or without a surveillance program integrated under 10 CFR part 50, appendix H."[70] If, however, "fewer than three surveillance data points exist for a specific material, then the embrittlement model must be used without performing the consistency check."[71]

In the event the embrittlement model deviates from the physical samples over the limits specified in the regulation, the licensee must submit additional evaluations and seek approval

[66] 10 C.F.R. § 50.61a(f)(6)(i)(B).

[67] 75 Fed. Reg. at 16 ("The NRC is modifying the final rule to include three statistical tests to determine the significance of the differences between heat-specific surveillance data and the embrittlement trend curve."). The consistency check compares the mean and slope of the embrittlement model curve against surveillance data, as well as checks to confirm that outliers fall within acceptable residual values provided in the regulation. See 10 C.F.R. § 50.61a(f)(6)(ii)–(v).

[68] 10 C.F.R. § 50.61a(f)(6)(i)(B).

[69] Id. § 50.61a(f)(6)(i)(A). Specifically, the regulation states, "[t]he surveillance material must be a heat-specific match for one or more of the materials" being evaluated through the embrittlement model. Id. The term "heat-specific," however, is not defined in the regulation or in the Alternate PTS Rule Technical Basis Report. The rulemaking, nonetheless, indicates that "heat-specific" refers to a material of the same composition as the type being modelled. See 75 Fed. Reg. at 16.

[70] 10 C.F.R. § 50.61a(a)(10) (emphasis added).

[71] Id. § 50.61a(f)(6)(i)(B).

for the deviations from the Director of the Office of Nuclear Reactor Regulation.[72] The rule, however, gives licensees some discretion in considering other plant-specific information that may be helpful in aligning their embrittlement models with the surveillance data.[73]

D. The Palisades LAR

Palisades submitted its LAR on July 29, 2014.[74] This appears to be the first instance in which a nuclear power plant licensee has requested a license amendment under the Alternate PTS Rule, 10 C.F.R. § 50.61a.[75] Palisades' LAR was accompanied by an "Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades" (Palisades Alternate PTS Rule Evaluation). It described the results of Entergy's evaluation of the Palisades RPV, pursuant to Section 50.61a.[76] It also provided Palisades' embrittlement model and RT_{MAX-X} calculations across various parts of the RPV,[77] the result of checks against surveillance data,[78] and an analysis of flaws in the RPV.[79]

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72 Id. § 50.61a(f)(6)(vi).

73 Id. § 50.61a(f)(6) ("The licensee shall verify that an appropriate RT_{MAX-X} value has been calculated for each reactor vessel beltline material by considering plant-specific information that could affect the use of the model . . . ."); 75 Fed. Reg. at 17 ("[T]he rule does not specify a method for adjusting the [model] value based on surveillance data, but rather requires the licensee to propose a case-specific [model] adjustment procedure . . . . [I]t is the NRC view that appropriate plant-specific adjustments based upon available surveillance data may be necessary to project reactor pressure vessel embrittlement for the purpose of this rule.").

74 See LAR.

75 See Tr. at 35.

76 Palisades Alternate PTS Rule Evaluation at v.

77 Id. §§ 3.1, 8.

78 Id. § 3.2.

79 Id. § 3.3.

The embrittlement model was checked against surveillance data for three different materials, one material representing the "base metal" for the Palisades RPV upper walls[80] and two materials representing different types of connecting welds.[81] Entergy acknowledged that its embrittlement model had to be checked against surveillance data for these three materials, because "the materials listed have at least three data points at three or more different neutron fluences," triggering the requirement to do a check under 10 C.F.R. § 50.61a(f)(6)(i)(B).[82]

The surveillance data representing the upper "base metal" of the RPV came from material samples taken directly from the Palisades RPV at different points in its operating life.[83] For the weld materials, there were not enough material samples pulled directly from the Palisades RPV to allow for a sufficient check.[84] Therefore, the surveillance data also "contain[ed] sister plant material data from H. B. Robinson Unit 2 (HB2), Indian Point Units 2 and 3 (IP2 and IP3), and Diablo Canyon Unit I (DCI)," other PWRs.[85] Entergy attested that these material samples from the sister plants were either of the same "Material Identification (Heat No.)," or same general type of material as the materials in the Palisades reactor.[86] The

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[80] Id. § 6 ("The base metal surveillance material is a heat-specific match for upper shell plate D-3802-1 and intermediate shell plates D-3803-1 and D-3803-3 (Heat C-1279).").

[81] Id. ("The weld wire surveillance materials are heat-specific matches for the upper, intermediate, and lower shell longitudinal welds (Heat W5214) and the intermediate to lower shell circumferential weld (Heat 27204).").

[82] Id. § 8.

[83] Id. tbl. 6-1.

[84] Compare 10 C.F.R. § 50.61a(f)(6)(i)(B) (requiring three or more samples to conduct a consistency check), with Palisades Alternate PTS Rule Evaluation tbls. 6-2 to 6-3 (only two samples from Palisades available for each weld material type analyzed).

[85] Palisades Alternate PTS Rule Evaluation § 6, tbls. 6-2 to 6-3.

[86] Id. tbl. 8.5.

fluence and reference temperatures shifts[87] for each material sample were provided to compare against the model.[88]

Entergy checked its embrittlement model against the surveillance data for the three material types for which such data was available,[89] and found that the results "satisfy the criteria in the Alternate PTS Rule."[90] Thus, Entergy concluded that its embrittlement model provided a satisfactory means to estimate RPV embrittlement under Section 50.61a.[91]

IV. Petitioners' Standing to Participate in this Proceeding

Entergy, but not the Staff, disputes Petitioners' standing. We conclude that Petitioners have satisfied the requirements for representational standing.

A. General Requirements for Standing

A petitioner's participation in a licensing proceeding requires a demonstration of standing. This requirement is derived from Section 189a of the Atomic Energy Act of 1954 (AEA),[92] which instructs the NRC to provide a hearing "upon the request of any person whose interest may be affected by the proceeding."[93] The Commission's regulation implementing the standing requirement, 10 C.F.R. § 2.309(d), directs a licensing board to consider (1) the nature of the petitioner's right under the AEA or the National Environmental Policy Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other

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[87] The reference temperature shift, or ΔT_{30}, is simply the difference in reference temperature from the unirradiated to the post-irradiated states. See id. § 3.1; see also 10 C.F.R. § 50.61a, equations 1, 5.

[88] Palisades Alternate PTS Rule Evaluation tbls. 6-2 to 6-3, 8-7 to 8-8.

[89] Id. § 8.2.

[90] Id.; see also id. § 9.

[91] Id. § 9.

[92] 42 U.S.C. § 2011 et seq.

[93] Id. § 2239(a)(1)(A); see also 10 C.F.R. § 2.105 (providing an opportunity for a hearing for "an amendment to an operating license, combined license, or manufacturing license").

interest in the proceeding; and (3) the possible effect of any decision or order that may be issued in the proceeding on the petitioner's interest.[94] When assessing whether an individual or organization has set forth a sufficient interest, the Commission has applied contemporaneous judicial concepts of standing, under which the petitioner must allege "a concrete and particularized injury that is fairly traceable to the challenged action and is likely to be redressed by a favorable decision."[95]

In certain circumstances, the Commission has adopted a proximity presumption that allows a petitioner living,[96] having frequent contacts,[97] or having a significant property interest[98] within fifty miles of a nuclear power reactor to establish standing without the need to make an individualized showing of injury, causation, and redressability.[99] The Commission has explained that the proximity presumption applies when there are "clear implications for the offsite environment, or major alterations to the facility with a clear potential for offsite consequences."[100] This impact can be assumed in such major actions as "construction permit and operating license proceedings for power reactors."[101] However, for the proximity

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[94] 10 C.F.R. § 2.309(d)(1)(ii)–(iv).

[95] Perry, CLI-93-21, 38 NRC at 92 (citing Lujan v. Defenders of Wildlife, 504 U.S. 555, 561 (1992)); see also, e.g., Yankee Atomic Electric Co. (Yankee Nuclear Power Station), CLI-98-21, 49 NRC 185, 195 (1998); Ga. Inst. of Tech. (Ga. Tech Research Reactor), CLI-95-12, 42 NRC 111, 115 (1995).

[96] Fla. Power & Light Co. (St. Lucie Nuclear Power Plant, Units 1 & 2), CLI-89-21, 30 NRC 325, 329 (1989) ("[L]iving within a specific distance from the plant is enough to confer standing on an individual or group in proceedings for construction permits, operating licenses, or significant amendments thereto.").

[97] Sequoyah Nuclear Fuels Corp. et al. (Gore, Okla. Site), CLI-94-12, 40 NRC 64, 75 (1994) (stating that the proximity presumption also applies to "persons who have frequent contacts in the area near a nuclear power plant").

[98] USEC, Inc. (Am. Centrifuge Plant), CLI-05-11, 61 NRC 309, 314 (2005).

[99] St. Lucie, CLI-89-21, 30 NRC at 329.

[100] See id.

[101] Id.

presumption to apply in the more limited license amendment proceedings, the proposed amendment must "'obvious[ly]' entail[] an increased potential for offsite consequences."[102]

Also, when, as here, an organization petitions to intervene in a proceeding, it must demonstrate either organizational or representational standing. To demonstrate organizational standing, the petitioner must show "injury-in-fact" to the interests of the organization itself.[103] When an organization seeks to establish representational standing, it must demonstrate that at least one of its members would be affected by the proceeding and identify that member. Moreover, the organization must show that the identified members would have standing to intervene in their own right, and that they have authorized the organization to request a hearing on their behalf.[104] In addition, the interests that the representative organization seeks to protect must be germane to its own purpose, and neither the asserted claim nor the relief sought must require an individual member to participate in the organization's legal action.[105]

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[102] Fla. Power & Light Co. (Turkey Point Nuclear Plant, Units 3 & 4), LBP-08-18, 68 NRC 533, 539 (2008) (first modification in original) (quoting Commonwealth Edison Co. (Zion Nuclear Power Station, Units 1 & 2), CLI-99-04, 49 NRC 185, 191 (1999) (internal quotation marks omitted) (quoting in turn St. Lucie, CLI-89-21, 30 NRC at 329–30)); see also Fla. Power & Light Co. (Turkey Point Nuclear Plant, Units 3 & 4), LBP-01-6, 53 NRC 138, 148 (2001) ("[T]he rule laid down in St. Lucie is intended to be applied across the board to all proceedings regardless of type because the rationale underlying the proximity presumption is not based on the type of proceeding per se but on whether 'the proposed action involves a significant source of radioactivity producing an obvious potential for offsite consequences.'" (quoting Ga. Tech, CLI-95-12, 42 NRC at 117)).

[103] See Calvert Cliffs 3 Nuclear Project, LLC et al. (Calvert Cliffs Nuclear Power Plant, Unit 3), CLI-09-20, 70 NRC 911, 915–16 (2009); Shaw Areva MOX Servs. (Mixed Oxide Fuel Fabrication Facility), LBP-07-14, 66 NRC 169, 183 (2007).

[104] See Gore, CLI-94-12, 40 NRC at 72 ("An organization seeking representational standing on behalf of its members may meet the 'injury-in-fact' requirement by demonstrating that at least one of its members, who has authorized the organization to represent his or her interest, will be injured by the possible outcome of the proceeding.") (citing Houston Lighting & Power Co. (Allens Creek Nuclear Generating Station, Unit 1), ALAB-535, 9 NRC 377, 389–400 (1979)).

[105] Consumers Energy Co. (Palisades Nuclear Power Plant), CLI-07-18, 65 NRC 399, 409 (2007).

B. Board Ruling on Petitioners' Standing

Each of the petitioning organizations seeks representational standing on behalf of one of its members.[106] The organizations explain that their members "seek to protect their lives, health and property by opposing the license amendment," and fear that the proposed amendment will lead to an increased risk of a loss-of-coolant accident because of a PTS event.[107] Petitioners argue that each individual has standing through the proximity presumption.[108] Petitioners argue that the proximity presumption applies in this proceeding as there is an obvious potential for offsite consequences generally in reactor operating license cases.[109]

Attached to the Amended Petition are affidavits of the four individual members. Bette Pierman is a member of Beyond Nuclear and resides approximately thirteen miles from Palisades.[110] Alice Hirt is a member of Don't Waste Michigan and resides approximately thirty-five miles from Palisades.[111] Maynard Kaufman is a member of Shoreline and resides approximately ten miles from Palisades.[112] Lastly, Gail Snyder is a member of NEIS, and, although she lives in Illinois, she owns five acres of land in Columbia, Michigan, approximately

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[106] Amended Petition at 1, 3, 5.

[107] Id. at 4.

[108] Id. at 3. The Petitioners add that "[a]ll of the petitioning individuals live within 50 miles of [Palisades]." Id.

[109] Id. at 3–4 (citing Pac. Gas & Electric Co. (Diablo Canyon Power Plant Independent Spent Fuel Storage Installation), LBP-02-23, 56 NRC 413, 426–27 (2002)).

[110] Amended Petition, attach., Amended Declaration of Bette Pierman in Support of Petition to Request a Public Hearing and Leave to Intervene in Opposition to Operating License Amendment for Palisades Nuclear Plant (Dec. 9, 2014).

[111] Amended Petition, attach., Declaration of Alice Hirt in Support of Petition to Request a Public Hearing and Leave to Intervene in Opposition to Operating License Amendment for Palisades Nuclear Plant (Dec. 1, 2014).

[112] Amended Petition, attach., Declaration of Maynard Kaufman in Support of Petition to Request a Public Hearing and Leave to Intervene in Opposition to Operating License Amendment for Palisades Nuclear Plant (Nov. 26, 2014).

fifteen miles from Palisades.[113] She further states that "[m]y family members have camped on the land, and go there during the warm season on day trips," and that she "lives, recreates and conducts business within the affected vicinity of the nuclear power plant."[114] She fears not only for her family's safety in the event of an accident at Palisades, but also that the land she owns "would become permanently uninhabitable."[115] Each of the individuals authorizes the petitioning organizations of which they are members to represent them in this proceeding.[116]

1. Proximity Presumption

Entergy opposes Petitioners' use of the proximity presumption, asserting that the Amended Petition lacks a specific, minimum demonstration that the license amendment portends an "obvious" potential for offsite consequences.[117] Entergy argues that Petitioners' reliance on Diablo Canyon is mistaken because Diablo Canyon concerned the licensing of an Independent Spent Fuel Storage Facility, while "Petitioners cite no authority for proximity-based standing in a license amendment proceeding similar to this one."[118] Entergy points to a 1998 Millstone decision, in which a licensing board declined to apply the proximity presumption, concluding that the proposed license amendment to add a safety-related sump pump subsystem to the existing system in the Engineered Safety Features building failed to present an obvious potential for offsite consequences.[119]

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[113] Amended Petition, attach., Declaration of Gail Snyder in Support of Petition to Request a Public Hearing and Leave to Intervene in Opposition to Operating License Amendment for Palisades Nuclear Plant (Nov. 30, 2014) [hereinafter Gail Snyder Affidavit].

[114] Id.

[115] Id.

[116] Supra notes 110–113.

[117] Entergy Answer at 12–15.

[118] Id. at 13.

[119] Id. at 13–14 (citing Ne. Nuclear Energy Co. (Millstone Nuclear Power Station, Unit No. 3), LBP-98-22, 48 NRC 149, 155, aff'd, CLI-98-20, 48 NRC 183, 184 (1998)).

The Staff disagrees with Entergy, concluding that the proximity presumption does apply to the proposed license amendment. The Staff argues that license amendments related to RPV embrittlement present an obvious potential for offsite public health and safety consequences.[120] Petitioners in their reply similarly argue "that a pressurized thermal shock-caused failure of a reactor pressure vessel raises an 'obvious potential for offsite consequences.'"[121] Petitioners also argue that the radius for the proximity presumption has to be at least as large as the range where obvious offsite consequences can occur.[122]

The Board finds that the proximity presumption applies to Petitioners. In Perry, cited by the Staff, a group of petitioners brought a contention concerning a license amendment to move "the schedule for the withdrawal of reactor vessel material specimens" from the technical specifications to the updated safety analysis report.[123] The petitioners argued that this move would limit their ability to challenge future amendments to the specimen withdrawal schedule.[124] The Commission concluded that "the instant amendment directly involves surveillance of the reactor vessel's integrity . . . . The material condition of the plant's reactor vessel obviously bears on the health and safety of those members of the public who reside in the plant's

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[120] NRC Staff Answer at 4 (quoting Perry, CLI-93-21, 38 NRC at 95–96).

[121] Reply at 15. Petitioners explain that, in St. Lucie, the Commission applied the proximity presumption even when the amendment only alleged "management's lack of the required character and competence," a less serious issue than alleged here. Id. (citing St. Lucie, CLI-89-21, 30 NRC 325)).

[122] See id. at 14–15 (citing Entergy Nuclear Operations and Entergy Nuclear Palisades, LLC (Palisades Nuclear Plant), CLI-08-19, 68 NRC 251, 254 (2008)). Petitioners argue that even if a reduced proximity presumption radius were to apply in this case, many of the petitioners live within ten to twenty miles of Palisades. Id. at 13–14 (citing Vt. Yankee Nuclear Power Corp. (Vt. Yankee Nuclear Power Station), LBP-87-7, 25 NRC 116, 118 (1987)).

[123] Perry, CLI-93-21, 38 NRC at 89.

[124] Id. at 90. According to the petitioners in Perry, "[i]f the license were amended, the public's only means to participate in future schedule changes would be through a request for action under 10 C.F.R. § 2.206," and the public would be unable to request a hearing in front of a licensing board. Id.

vicinity."[125] The Commission determined that the petitioners had standing even though they did not provide a reactor vessel failure scenario.[126]

Petitioners' contention relates to a similar potential injury, a release of radiation due to the potential failure of RPV integrity. It is obvious to this Board, as it was to the Commission in Perry, that a change in the safety-related requirements intended to ensure the integrity of the RPV "obviously bears on the health and safety of those members of the public who reside in the plant's vicinity."[127] That is all the more apparent in this case because, as Entergy acknowledges, the alternative regulatory requirements proposed by the license amendment are less conservative than those that the amendment is intended to replace.[128]

Entergy's reliance on the licensing board decision in Millstone is misplaced.[129] The licensing board in that case was understandably confounded by the petitioner's challenge to the addition of a safety system: "[E]ven assuming the instant amendment to add a safety-related sump pump subsystem to the existing sump pump system . . . somehow presents the potential for offsite environmental consequences, that potential is anything but obvious."[130] The circumstances in Millstone are entirely different from those here, where the potential for offsite consequences from a failure of RPV integrity is obvious.[131]

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[125] Id. at 95–96.

[126] Id. at 95.

[127] Id. at 95–96; see also Turkey Point, LBP-01-6, 53 NRC at 149–50 (stating that licensing actions that potentially increase reactor vessel embrittlement, such as license renewals, "hold the potential for offsite consequences that are obvious").

[128] See Entergy Answer at 6–7.

[129] Millstone, LBP-98-22, 48 NRC at 149.

[130] Id. at 155.

[131] Entergy also alleges that the Amended Petition merely repeats arguments from a prior Palisades license renewal proceeding and is not specific to the license amendment at issue. Entergy Answer at 14. The Board disagrees with Entergy. The Amended Petition presents a specific argument geared towards the LAR. See Amended Petition at 11–22; cf. Millstone, LBP-

2. NEIS's Standing

Entergy separately challenges NEIS's standing, alleging that Ms. Snyder, the NEIS member petitioning to intervene, fails to meet the proximity presumption in her own right.[132] Entergy argues that Ms. Snyder's affidavit is not sufficiently specific to show frequent contact within fifty miles of Palisades,[133] as it does not provide an address for her property or give the duration of her family members' visits.[134] Furthermore, Entergy asserts that Ms. Snyder cannot request standing on the basis of third parties, given that "Ms. Snyder's declaration does not claim that she ever visits her property."[135]

The Staff maintains, however, that Ms. Snyder demonstrates standing. Although the Staff agrees with Entergy that she may not be able to claim standing based on her family's activities or the frequency of her own contacts, it notes that a "harm to a property interest is also sufficient to establish standing."[136] The Staff acknowledges Ms. Snyder's concern that her property could become uninhabitable in the event of an accident at Palisades.[137] Petitioners reply that Ms. Snyder also "camps and picnics" on the property she owns.[138]

98-22, 48 NRC at 155–56 (A petition was not sufficiently specific when it "merely repeat[ed] the contents of [the petitioner's] earlier petition" concerning a prior license amendment.).

[132] Entergy Answer at 15.

[133] Id. at 15–16 (citing Bell Bend, CLI-10-07, 71 NRC at 136, 140).

[134] Id. at 16.

[135] Id. Entergy cites as support a licensing board decision in Fermi, in which a mother was allegedly denied standing based on her son's residence within fifty miles of a power plant, because she herself lived more than fifty miles away. See id. at 16–17 (citing Detroit Edison Co. (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-470, 7 NRC 473, 474 n.1 (1978)).

[136] NRC Staff Answer at 5 n.17 (citing Am. Centrifuge, CLI-05-11, 61 NRC at 314).

[137] Id.

[138] Reply at 14.

Entergy is correct that Ms. Snyder cannot gain standing from the interests of third parties except in very limited circumstances not present here.[139] Moreover, to demonstrate "frequent contacts" within the fifty mile site radius under the proximity presumption, Ms. Snyder must show that her contacts are "substantial" and "regular," and must describe them with specificity.[140] Although Ms. Snyder's affidavit indicates she may spend time by the Palisades site,[141] these statements are too vague to demonstrate a substantial or regular presence within fifty miles of Palisades.[142]

Nonetheless, the Staff is correct that a property interest is sufficient to grant standing based on proximity. As the Commission noted in American Centrifuge, "[t]he Atomic Energy Act authorizes the Commission 'to accord protection from radiological injury to both health and *property* interests.' Thus, a genuine property interest . . . is sufficient to accord [the petitioner] standing, given that the home is located" within close proximity to the facility.[143] Ms. Snyder has clearly enunciated her concern that an accident at Palisades could render her five acres of land

[139] See St. Lucie, CLI-89-21, 30 NRC at 329 (A petitioner "may not derive standing from the interests of another person or organization"); see also Fermi, ALAB-470, 7 NRC at 474 n.1 (noting that a parent could attain standing through reference to her child if the child was "a minor or otherwise under a legal disability," and thus unable to participate herself); Nuclear Fuel Servs. (Erwin, Tenn.), LBP-04-5, 59 NRC 186, 193 n.10 (A petitioner could not rely on "caretakers [] maintaining and farming the property in [the petitioner's] absence" as grounds for standing.), aff'd, CLI-04-13, 59 NRC 244 (2004).

[140] See PPL Bell Bend, LLC (Bell Bend Nuclear Power Plant), CLI-10-07, 71 NRC 133, 140 (2010). This is a determination to be made by a licensing board after weighing all the information provided. See id. at 139.

[141] See Gail Snyder Affidavit (stating that she "lives, recreates and conducts business" within the vicinity of the plant).

[142] See Bell Bend, CLI-10-07, 71 NRC at 140 (The Commission concluded that the petitioner's statement that he "routinely pierces the 50-mile proximate rule [sic] during his day-to-day activities" by itself was "too vague a statement on which to base standing.").

[143] Am. Centrifuge, 61 NRC at 314 (quoting Gulf States Utilities Co. (River Bend Station, Unit 1) CLI-94-10, 40 NRC 43, 48 (1994) (citing 42 U.S.C. §§ 2133(b), 2201(b))) (footnote omitted).

"permanently uninhabitable."[144] The Board thus finds that she has demonstrated a sufficient property interest to warrant standing based on proximity.

3. Representational Standing

Neither Entergy nor the Staff challenge Petitioners' request for representational standing. Although the Board has the obligation to independently assess Petitioners' standing,[145] we have no difficulty concluding that the requirements for representational standing are met in this case. As discussed above, Petitioners have provided affidavits from their members, each of whom has standing under the proximity presumption and has authorized Petitioners to request a hearing on their behalf.[146] Petitioners have also demonstrated that the interests the representative organizations seek to protect are germane to their own purposes, and that neither the asserted claims nor the relief sought require an individual member to participate in the organization's legal action.[147]

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[144] Gail Snyder Affidavit. Entergy faults Ms. Snyder for not listing the address of her land in her affidavit. Entergy Answer at 16. However, she has stated that the property is located in Columbia, Michigan, and that it is located approximately fifteen miles from Palisades. Gail Snyder Affidavit. Given that Entergy does not question whether the property actually exists, or whether she owns it, we do not find the failure to provide an exact address in her affidavit a limiting concern. See Am. Centrifuge, 61 NRC at 314–15 (the Commission examined whether the petitioner actually owned the property only after the licensee challenged ownership in its answer).

[145] See 10 C.F.R. § 2.309(d)(2); supra notes 110–113 and accompanying text.

[146] Gore, CLI-94-12, 40 NRC at 72.

[147] See Declaration of Authorized Officer of Beyond Nuclear in Support of Petition to Intervene in Docket No. 50-255 (Dec. 1, 2014); Declaration of Authorized Officer of Don't Waste Michigan in Support of Petition to Intervene in Docket No. 50-255 (Dec. 1, 2014); Declaration of Authorized Officer of Michigan Safe Energy Future in Support of Petition to Intervene in Docket No. 50-255 (Dec. 1, 2014); Declaration of Authorized Officer of Nuclear Energy Information Service in Support of Petition to Intervene in Docket No. 50-255 (Dec. 1, 2014); see also Palisades, CLI-07-18, 65 NRC at 409.

V. Admissibility of Petitioners' Contention

A. General Pleading Requirements

In order to participate as a party in this proceeding, a petitioner for intervention must not only establish standing, but must also proffer at least one admissible contention that meets the requirements of 10 C.F.R. § 2.309(f).[148] An admissible contention must: (i) provide a specific statement of the legal or factual issue sought to be raised; (ii) provide a brief explanation of the basis for the contention; (iii) demonstrate that the issue raised is within the scope of the proceeding; (iv) demonstrate that the issue raised is material to the findings the NRC must make to support the action that is involved in the proceeding; (v) provide a concise statement of the alleged facts or expert opinions, including references to specific sources and documents, that support the petitioner's position and upon which the petitioner intends to rely at the hearing; and (vi) provide sufficient information to show that a genuine dispute exists in regard to a material issue of law or fact, including references to specific portions of the application that the petitioner disputes, or, in the case when the application is alleged to be deficient, the identification of such deficiencies and supporting reasons for this belief.[149]

The purpose of Section 2.309(f)(1) is to "focus litigation on concrete issues and result in a clearer and more focused record for decision."[150] The Commission has stated that it "should not have to expend resources to support the hearing process unless there is an issue that is appropriate for, and susceptible to, resolution in an NRC hearing."[151] The rules on contention admissibility are "strict by design."[152] Petitioners must comply with all of these requirements.

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[148] See 10 C.F.R. § 2.309(a).

[149] Id. § 2.309(f)(1).

[150] Changes to Adjudicatory Process, 69 Fed. Reg. 2182, 2202 (Jan. 14, 2004).

[151] Id.

[152] See, e.g., Dominion Nuclear Conn., Inc. (Millstone Nuclear Power Station, Unit 2), CLI-03-14, 58 NRC 207, 213 (2003); Dominion Nuclear Conn., Inc. (Millstone Nuclear Power Station, Units

B. Scope of Review of License Amendments

The NRC regulations define the Commission's scope of review of a license amendment application broadly: "In determining whether an amendment to a license, construction permit, or early site permit will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses, construction permits, or early site permits to the extent applicable and appropriate."[153] The "considerations" the Commission should review include those defined in 10 C.F.R. § 50.40, titled "Common standards." As the Atomic Safety and Licensing Appeals Board explained:

> In essence, Section 50.40 requires that the Commission be persuaded, inter alia, that the applicant will comply with all applicable regulations, that the health and safety of the public will not be endangered, that the issuance of the amendment will not be inimical to the health and safety of the public, and that any applicable requirements of 10 CFR Part 51 (governing environmental protection) have been satisfied.[154]

C. Prohibition Against Challenging NRC Rules in Agency Adjudications

The NRC's adjudicatory process is not the venue for challenging the NRC's regulations. When the Commission has opted to address a safety or environmental concern through regulation, it has uniformly prohibited litigation of that same issue in a site-specific adjudicatory proceeding: "Contentions that are the subject of general rulemaking by the Commission may not

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2 & 3), CLI-01-24, 54 NRC 349, 358–59 (2001); Duke Energy Corp. (Oconee Nuclear Station, Units 1, 2, & 3), CLI-99-11, 49 NRC 328, 334–35 (1999).

[153] 10 C.F.R. § 50.92(a).

[154] N. States Power Co. (Prairie Island Nuclear Generation Plant, Units 1 and 2) et al., ALAB-455, 7 NRC 41, 44 (1978); see also Tenn. Valley Auth. (Browns Ferry Nuclear Plant, Units 1, 2, & 3), ALAB-664, 15 NRC 1, 15–16 ("Prior to license issuance the NRC must first find reasonable assurance that the activities authorized by the amendment can be conducted without endangering the health and safety of the public, and in compliance with Commission regulations."), vacated and remanded on other grounds, CLI-82-26, 16 NRC 880 (1982); Fla. Power & Light Co. (Turkey Point Nuclear Generating Station, Units 3 & 4), LBP-81-16, 13 NRC 1115, 1120 (1981) (reviewing a proposed license amendment to determine whether it would "endanger the health and safety of the public").

be litigated in individual license proceedings."[155] According to 10 C.F.R. § 2.335(a), "no rule or regulation of the Commission, or any provision thereof . . . is subject to attack" in an adjudicatory proceeding unless a waiver is granted by the Commission.[156]

D. Board Ruling on Contention Admissibility

Petitioners claim that Entergy's LAR "deviate[s] from the regulatory requirements of 10 C.F.R. § 50.61 to §50.61a (Alternate Fracture Toughness Requirements)."[157] They assert that "Palisades has an acknowledged problem of worsening reactor vessel embrittlement commencing from the start of operations in the early 1970's," and "[b]asically, 10 C.F.R. § 50.61a allows Entergy to substitute various estimates of the status of the RPV for actual data investigation and analysis."[158] Petitioners "further raise the question of whether Entergy should be allowed to resort to § 50.61a at all."[159]

Petitioners provide three specific bases for their contention:

1. **"Analytical vs. Experimental."**[160] Petitioners argue that Entergy cannot provide reasonable assurance of public health and safety under the Alternate PTS Rule without obtaining or using additional data from the Palisades RPV.
2. **"The Comparable Plants Are Not Apples-to-Apples Comparisons."**[161] Petitioners argue that "sister plant" surveillance data from reactors with different

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[155] Calvert Cliffs 3 Nuclear Project, LLC (Calvert Cliffs Nuclear Power Plant, Unit 3) et al., CLI-14-8, 80 NRC 71, 79 n.27 (2014) (citing Oconee, CLI-99-11, 49 NRC at 345).

[156] 10 C.F.R. § 2.335(a). A party can petition for a waiver of a specific NRC regulation, based on a showing of "special circumstances" such that application of the rule would not serve the purposes for which it was adopted. Id. § 2.335(b); see also Dominion Nuclear Conn., Inc. (Millstone Nuclear Power Station, Units 2 & 3), CLI-05-24, 62 NRC 551, 559–60 (2005) (laying out a four-factor test for determining whether to grant a waiver). However, as Petitioners have not petitioned for a waiver of any NRC regulation, this process need not be discussed further.

[157] Amended Petition at 11–12.

[158] Id. at 10.

[159] Id. at 11.

[160] Id. at 15.

[161] Id. at 16.

operating characteristics cannot be combined with Palisades' surveillance data for purposes of the Section 50.61a(f)(6) consistency check.

3. **"Cross-Comparisons And Standard Deviations Don't Match Up."**[162] Petitioners argue the applicant's use of surveillance data does not account for spatial variability in fluence across a reactor, and that this variability increases beyond regulatory limits when sister plant surveillance data is used.

Petitioners apparently want the Board to preclude Entergy from relying on Section 50.61a to avoid meeting the requirements of Section 50.61, but it is just such a "deviation" that Section 50.61a authorizes. The evident purpose of the Alternate PTS Rule's "Alternate Fracture Toughness Requirements" is to provide an alternative to satisfying the more demanding requirements of Section 50.61. Therefore, Petitioners are in substance asking that the Board prohibit what Section 50.61a allows. Under 10 C.F.R. § 2.335, we may not consider such a contention except under specific conditions not present here.[163]

Nevertheless, because the petition provides three potential bases of the contention,[164] each of which might be able to stand alone as a separate contention, we have reviewed each of the asserted bases to determine whether any could satisfy the contention admissibility requirements in Section 2.309(f)(1) and also comply with Section 2.335's prohibition on

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[162] Id. at 18.

[163] See supra note 156.

[164] The petition includes a fourth basis, which argues that Entergy's equivalent margins analysis allows Palisades to operate its RPV outside of permissible limits. Amended Petition at 19. In their reply, Petitioners appear to agree with Entergy and the NRC Staff that the equivalent margins analysis is actually the subject of a separate license amendment request. Reply at 11–12. Petitioners have, since filing the petition in this case, filed a separate petition challenging Entergy's separate license amendment request to authorize the equivalent margins analysis. See Petition to Intervene and for a Public Adjudication Hearing of Entergy License Amendment Request for Approval of 10 CFR Part 50 Appendix G Equivalent Margins Analysis, Docket No. 50-255-LA2 (Mar. 9, 2015). A licensing board has been appointed for that separate proceeding. See Entergy Nuclear Operations, Inc., Establishment of Atomic Safety and Licensing Board, 80 Fed. Reg. 15,827 (Mar. 25, 2015); Commission Order (Establishment of Atomic Safety and Licensing Board), Docket No. 50-255-LA2 (Mar. 19, 2015) (unpublished). Petitioners' challenge to the equivalent margins analysis license amendment request is pending before that board. This Board will therefore not consider further the fourth asserted basis of the contention.

challenging agency regulations.[165] We conclude that none of the asserted bases could satisfy both requirements.

1. Basis 1: Use of Analytical Models Rather than Empirical Data

Petitioners contend that the Entergy LAR fails to ensure public health and safety because the analyses undergirding the LAR estimate current and future embrittlement of the Palisades RPV without reliance on empirical data from material samples.[166] The last material sample taken from the Palisades RPV to measure embrittlement was removed in 2003, while the next sample is not scheduled to be removed until 2019. Thus, Petitioners emphasize, "fully 16 years will have passed without development or analysis of new physical evidence of embrittlement."[167] Quoting from the Declaration of Arnold Gunderson, a nuclear engineer, Petitioners argue that "the NRC has allowed Palisades to make unrealistic, unsupported and imprudent safety calculations based on little more than probabilistic risk."[168]

As an alleged example of the dangers of ignoring physical data in favor of modeling, Petitioners claim that a material sample, capsule A-60, was deleted from the Palisades Surveillance Program back in 1984 "precisely because it gave an answer that would have

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[165] Licensing Boards have the authority to reformulate contentions "to consolidate issues for a more efficient proceeding." Crow Butte Res., Inc. (N. Trend Expansion Project), CLI-09-12, 69 NRC 535, 552 (2009) (quoting Shaw Areva MOX Servs. (Mixed Oxide Fuel Fabrication Facility), LBP-08-11, 67 NRC 460, 482 (2008)) (internal quotation marks omitted).

[166] Amended Petition at 14–16. According to Gundersen, "[a]nalysis is no replacement for testing the capsule coupon." Gundersen Declaration ¶ 55.

[167] Amended Petition at 15.

[168] Id. at 16 (quoting Gundersen Declaration ¶ 23) (internal quotation marks omitted); see also Gundersen Declaration ¶ 20 (citing Palisades Webinar at 1, encl. 2 at 6 (discussing the surveillance data removal schedule for the Palisades facility)). Gundersen further claims that the agency should have adopted a more evidence-driven approach, but instead has consistently acted otherwise for economic reasons. Gundersen Declaration ¶¶ 16, 24.3.

required Palisades to be shut down."[169] Petitioners repeated at oral argument that Palisades' LAR "ignores" the data from the alleged 1984 testing of capsule A-60.[170]

Entergy responds that this contention is a challenge to Section 50.61a itself, which is impermissible under 10 C.F.R. § 2.335.[171] The Staff's answer also emphasizes that Section 50.61a is a Commission rule and thus a "petitioner cannot simply argue that § 50.61a is flawed because it fails to require an applicant to do X or should not allow an applicant to do Y."[172]

Regarding the material sample that was allegedly discarded in 1984, capsule A-60, Entergy responds that it did not discard unfavorable data, but instead simply discarded a sample that was "inadvertently over-irradiated in the 1980s."[173] At oral argument Entergy claimed that this capsule was never tested, and thus cannot provide any evidence of embrittlement trends useful for Palisades' LAR.[174] In its answer, the Staff states that an identical capsule, capsule A-240, was placed in a diametrically opposite position with similar neutron fluences and temperatures as capsule A-60, "making withdrawal and testing of Capsule A-60 unnecessary."[175] Both Entergy and the Staff also emphasize that Petitioners are complaining about a separate agency action that occurred in 1984, well outside the scope of this proceeding.[176]

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[169] Amended Petition at 19 (quoting Gundersen Declaration ¶ 42) (internal quotation marks omitted).

[170] Tr. at 15, 65.

[171] Entergy Answer at 21.

[172] NRC Staff Answer at 16–17.

[173] Entergy Answer at 31 n.160. At oral argument, Entergy clarified that "there was an outage when it was scheduled to be removed and they had difficulty removing it. . . . So, they had to leave it in for another cycle. And when eventually they did remove it, it had experienced more irradiation than it would have experienced even beyond 80 years of plant [operation]." Tr. at 93.

[174] Tr. at 85, 95, 96, 118.

[175] NRC Staff Answer at 27 n.123.

[176] Entergy Answer at 31 n.160; NRC Staff Answer at 27.

In their reply, Petitioners reassert that their claims are not impermissible attacks on NRC regulations.[177] Petitioners argue that "their expert's critique of the means by which the § 50.61a investigation was conducted . . . cannot be construed as a frontal assault on the regulatory citadel, but must instead be seen, for purposes of the admissibility determination, as an exposé of the flaws caused by straying away from knowable science."[178] Petitioners comment that their concerns about capsule A-60 are not irrelevant legal arguments, but are instead evidentiary observations, which allegedly show "that the degree of RPV embrittlement in the 1980's was greatly advanced, given the then-short operational age of the reactor."[179] At oral argument Petitioners noted that "[t]here is some seriously conflicting information about the status of the [A-60] capsule,"[180] and asserted that there is evidence showing, contrary to Entergy's claim, that capsule A-60 was indeed tested and embrittlement data noted.[181]

The Board agrees with Entergy and the Staff that Petitioners' general claims concerning the use of analytical model results over physical data do not lead to an admissible contention because they amount to a challenge to the Alternate PTS Rule. The Commission noted when it promulgated Section 50.61a that this rule "provides reasonable assurance" of public health and safety, thereby endorsing the 50.61a embrittlement model approach and precluding requests to create requirements more restrictive than the rule.[182] As Entergy correctly states, "[w]hen a Commission regulation permits the use of a particular analysis, a contention asserting that a

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[177] Reply at 3.

[178] Id. at 4–5.

[179] Id. at 10–11.

[180] Tr. at 128.

[181] Id. at 129–30.

[182] 75 Fed. Reg. at 22.

different analysis or technique should be utilized is inadmissible because it indirectly attacks the Commission's regulations."[183]

As Intervenors note, although material samples had been pulled from the RPV at a relatively consistent three to five year interval since the reactor became operational,[184] there is now a projected sixteen year gap between the removal of the last sample, capsule W-100, in 2003, and the pulling of the next sample in 2019.[185] But, by advocating that the Board require the testing of additional samples, Intervenors are asking the Board to demand more than Section 50.61a requires.

We are also not persuaded by Petitioners' claim that Palisades' LAR "ignores" the data from the alleged 1984 testing of capsule A-60.[186] Section 50.61a defines surveillance data broadly, to include "any data that demonstrates the embrittlement trends for the beltline materials."[187] If the capsule had in fact been tested, the resulting data could constitute surveillance data relevant to evaluating embrittlement trends.[188] Entergy appeared to

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[183] Entergy Answer at 22 (quoting Detroit Edison Co. (Fermi Nuclear Power Plant, Unit 3), LBP-09-16, 70 NRC 227, 255 (internal quotation marks omitted) (citing Metro. Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), LBP-83-76, 18 NRC 1266, 1273 (1983)), aff'd on other grounds, CLI-09-22, 70 NRC 932, 933 (2009)).

[184] Office of Nuclear Reactor Regulation, Approval of Proposed Reactor Vessel Surveillance Capsule Withdrawal Schedule (Aug. 14, 2007), encl., Neil K. Ray, Surveillance Capsule Withdrawal Schedule, Palisades Nuclear Plant at 3 (ADAMS Accession No. ML071640310) (listing the capsule removal schedule); Palisades Webinar, encl. 2 at 6 (indicating that an additional capsule, capsule SA-240-1, was removed approximately three years prior to removal of capsule W-100 in 2003).

[185] Palisades Webinar at 1, encl. 2 at 6.

[186] Tr. at 15, 65.

[187] 10 C.F.R. § 50.61a(a)(10).

[188] Entergy and the NRC Staff give slightly different reasons in their answers for why the capsule was removed from the program. Compare Entergy Answer at 31 n.160 (claiming the capsule was "accidentally over-irradiated in the 1980s") with NRC Staff Answer at 27–28, 28 n.123 ("[B]ecause [equivalent] Capsule A-240 had been withdrawn and tested, it could be used to predict the end-of-life material properties of the Palisades reactor vessel, making withdrawal and testing of Capsule A-60 unnecessary.").

acknowledge that much at oral argument.[189] Thus, a contention alleging that Entergy should have evaluated surveillance data actually obtained from capsule A-60 would not violate Section 2.335's prohibition on contentions challenging agency regulations.

Nonetheless, Intervenors have not provided any factual support for their assertion that this capsule was indeed removed and tested for embrittlement data.[190] As noted above, Mr. Gunderson claims that capsule A-60 was deleted from the Palisades Surveillance Program in 1984 "precisely because it gave an answer that would have required Palisades to be shut down."[191] Mr. Gunderson fails to explain, however, how he deduced that the capsule was tested, much less how he knows that the testing produced such a significant result. Although the contention admissibility stage is not the appropriate point at which to evaluate witness credibility or to weigh competing evidence, an expert must provide a reasoned basis or explanation for opinions in support of a contention.[192] Mr. Gunderson has provided no such basis or explanation for his belief that the capsule was tested approximately thirty years ago and that the results would have required Palisades to shut down. Amendment 79 to the Palisades license, which authorized the removal of capsule A-60, does not provide any support for Mr. Gunderson's assertions.[193] In the absence of any factual support for Petitioners' argument, capsule A-60 is not "data that demonstrates the embrittlement trends for the beltline material,"[194] and therefore the fact that it was excluded from the Palisades LAR is not a material issue.

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[189] See Tr. at 101.

[190] Id. at 85, 95, 96, 118.

[191] Gundersen Declaration ¶ 42.

[192] USEC, Inc. (Am. Centrifuge Plant), CLI-06-10, 63 NRC 451, 472 (2006).

[193] See Office of Nuclear Reactor Regulation, Safety Evaluation Report Supporting Amendment 79 to Consumers Power Company Provisional Operating License at 2 (Feb. 28, 1984) (ADAMS Accession No. ML020800206).

[194] 10 C.F.R. § 50.61a(a)(10).

2. Bases 2 & 3: Use of Sister Plant Comparison Data

We will discuss Bases 2 and 3 together because both concern Entergy's consistency check of its embrittlement model under Section 50.61a(f)(6), and the use of sister plant surveillance data as part of that check.

Under Basis 2, Petitioners contend that the surveillance data provided from other PWR reactor vessels at H. B. Robinson, Indian Point, and Diablo Canyon cannot be compared with the material samples from the Palisades RPV for purposes of verifying the embrittlement model.[195] Petitioners' expert, Mr. Gundersen, states that "[w]hile it is true that the material used to weld the reactor plates together to create the reactor vessel is similar among the four plants, the dramatically different nuclear core design and operational power characteristics make an accurate comparison impossible."[196] According to Mr. Gundersen, the different core design and operational characteristics of these reactors are relevant because they "impact[] the neutron flux on each reactor vessel, thus making an accurate comparison of neutron bombardment and embrittlement impossible," with the Palisades embrittlement model.[197] According to Petitioners, the Staff acknowledges that use of "all possible" plant-specific surveillance data is critical for an effective check of an embrittlement model.[198]

Petitioners' discussion under Basis 3 offers two more specific lines of argument regarding the use of sister plant data. First, Petitioners claim that "there is extraordinary

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[195] Amended Petition at 16.

[196] Gundersen Declaration ¶ 27.

[197] Id.

[198] Amended Petition at 20–21 (citing Gundersen Declaration ¶ 53); see also Transcript, 619th Meeting, Advisory Committee on Reactor Safeguards (Nov. 6, 2014) (ADAMS Accession No. ML14321A542). Although Petitioners initially state that this view came from the Advisory Committee on Reactor Safeguards, they later assert that it instead came from Mark Kirk from the Office of Regulatory Research, the alleged "primary author of § 50.601a." Reply at 7. Petitioners in their reply also cite to minutes of a meeting of the Advisory Committee on Reactor Safeguards, alleging that parts of the agency itself have a "dark" and unsafe view of Palisades' RPV embrittlement. Id. at 16.

[spatial] variability between the neutron flux across the nuclear core in" the Palisades reactor.[199] In his declaration, Mr. Gundersen contends that given this spatial variability, it is impossible to compare multiple samples from multiple reactors to derive the flux or fluence for a single specific area of an RPV without introducing error.[200] For support, Mr. Gundersen cites to a Palisades Reactor Pressure Vessel Fluence Evaluation (Palisades Fluence Evaluation Report), conducted in 2011, which describes "the methodology used in the fluence evaluations for the Palisades plant."[201] He points specifically to two charts that allegedly show how neutron flux and fluence vary across the Palisades reactor over location and over time.[202] He cites an additional 1990 report, which allegedly concludes that a number of factors, including RPV dimensions and cycle variations, can cause fluence at an RPV wall to vary up to 25% from predictions.[203]

Second, Petitioners claim that "the most serious analytical problem in the use of sister plants" is the alleged difficulty or impossibility of the data from the sister plants staying within

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[199] Amended Petition at 18 (quoting Gundersen Declaration ¶ 34) (internal quotation marks omitted).

[200] Gundersen Declaration ¶¶ 30–38.

[201] Id. ¶ 30 (citing Westinghouse, Palisades Reactor Pressure Vessel Fluence Evaluation, WCAP-15353, Supplement 2, Revision 0 (July 2011) (ADAMS Accession No. ML14316A207) [hereinafter Palisades Fluence Evaluation Report]).

[202] See id. ¶¶ 34, 35. Both charts cited by Mr. Gundersen (charts 2.2-3 and 2.2-4) actually discuss flux characteristics for the Palisades RPV; neither discusses fluence. At oral argument, Petitioners clarified that Mr. Gundersen meant to cite to chart 2.2-5, which discusses fluence, instead of chart 2.2-4. Tr. at 127. Petitioners also commented that the two concepts are related, as "flux essentially drives fluency." Id. at 59.

[203] Gundersen Declaration ¶ 37 (citing Analysis of the Reactor Pressure Vessel Fast Neutron Fluence and Pressurized Thermal Shock Reference Temperatures for The Palisades Nuclear Plant § 4.3 (May 17, 1990) (ADAMS Accession No. ML052720270) (page 55 of PDF counter) [hereinafter Palisades 1990 Fluence Analysis] (noting that a number of factors contribute to fluence uncertainty along a reactor vessel wall, including material composition, vessel dimensions, and cycle-by-cycle variation)).

one standard deviation, or 20%, from Palisades' data.[204] Petitioners, however, do not point to any regulation as the origin of this alleged requirement. Instead, Mr. Gundersen cites the Palisades Fluence Evaluation Report, which states that "the [fluence] calculations and [material sample] measurements should agree within 20% at the 1σ level."[205] Mr. Gundersen claims that, as a result, "[a] 1σ analysis appears to be binding within the Palisades data," but the NRC has allowed the use of sister plant data "without requiring the same 1σ variance with Palisades."[206]

Entergy responds that Bases 2 and 3 of the contention are vague, unclear, and "do not articulate any specific deficiency in Entergy's LAR."[207] Entergy admits that sister plant data is required for consistency checks of the embrittlement model,[208] but contends that Section 50.61a(f)(6) requires the use of any available sister plant data, if it is of the same material heats used in the Palisades RPV.[209] Entergy asserts it thus had to use the sister plant data, and any allegations it should not have done so are challenges to the Commission's rules.[210] Turning to Petitioners' suggestion that there is a hard 20% deviation limit among material samples, Entergy

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[204] Amended Petition at 18. Petitioners do not specify the specific variable to which this 20% limit applies.

[205] Gundersen Declaration ¶ 30. In his declaration, Gundersen also alleges that the NRC authorized deletion of capsule A-60 "because its measured neutron value exceeded this 1σ variation." See id. ¶¶ 40–42 (citing Division of Licensing, Amendment to Provisional Operating License, Amendment No. 79 to License No. DPR-20, Reactor Vessel Surveillance Capsule Program (Feb. 28, 1984) (ADAMS Accession No. ML020800206)).

[206] Id. ¶ 32.

[207] Entergy Answer at 25. Entergy also noted at oral argument that "neither the Petition nor the Gundersen declaration makes a single reference to the actual Westinghouse report [Palisades Alternate PTS Rule Evaluation] that was submitted with the license application to show compliance with Section 50.61a." Tr. at 70.

[208] Entergy Answer at 27.

[209] Id.

[210] Id. ("Entergy had no discretion here: 'If three or more surveillance data points measured at three or more different neutron fluences *exist for a specific material*, the licensee *shall* determine if the surveillance data show a significantly different trend than the embrittlement model predicts.'" (quoting 10 C.F.R. § 50.61a(f)(6))).

responds that Petitioners mix "apples and oranges in multiple ways."[211] Referring to Regulatory Guidance 1.190, used for calculating neutron fluence in a reactor vessel,[212] Entergy maintains that "[t]he 1σ/20% standard . . . applies to estimates of the *uncertainty* in specific fluence calculations at a particular location—not to 'variations' in fluence across the core at different locations."[213] Entergy also maintains that "[t]he 20%/1 σ screening standard for plant-specific fluence inputs" is not relevant to any of the consistency checks using sister plant data performed in accordance with Section 50.61a.[214]

The Staff argues that "surveillance data" is necessarily a broad term, and a consistency check with sister plant data is required whenever there is a "heat-specific" match and three or more data points exist.[215] The Staff asserts that "Petitioners have not challenged Entergy's compliance with the rule by alleging, for example, that the surveillance data from sister plants is not a 'heat-specific match' or that Entergy's analysis of the sister-plant data was deficient," but instead have challenged the concept of using sister-plant data at all.[216] Regarding Mr. Gundersen's concerns about variability in neutron flux among sister plant reactor cores and the existence of a 20% limit, the Staff generally responds that this contention does "not identify any error or omission in Entergy's LAR analysis," or otherwise indicate any flaws in the application.[217]

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[211] Id. at 29.

[212] Office of Nuclear Regulatory Research, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, Regulatory Guide 1.190 (Mar. 2001) (ADAMS Accession No. ML010890301) [hereinafter Regulatory Guide 1.190].

[213] Entergy Answer at 31.

[214] Id. at 29–30. Entergy admits, however, that the 20% requirement applies when using calculated fluence values to project "the reference transition temperature for a material." Id.

[215] NRC Staff Answer at 21–22 (citing 10 C.F.R. § 50.61a(a)(10)).

[216] See id. at 22–24.

[217] Id. at 26.

Petitioners reply that they do not mean to argue "that Entergy should not be allowed to analyze sister-plant data at all,"[218] but that "comparables should be comparable," and Entergy misapplied the consistency check in this instance.[219] Petitioners state that Mr. Gundersen critiqued the specific use of surveillance samples from the other plants, but did not attack the surveillance data prong of the Alternate PTS Rule as a whole.[220] Petitioners argue that under Section 50.61a there is significant leeway in the calculation and verification of RT_{MAX-X} under the Alternate PTS Rule, and the methodology and choice of data for those analyses must be disclosed and reviewed.[221] According to Petitioners, "[w]here there is discretion vested in the regulator, differences of opinion, interpretation, and expert analysis are legitimate bases for challenging the decision."[222]

Petitioners also contend in their reply that even if their contention goes beyond the bounds of what is covered under Section 50.61a, "a contention about a matter not covered by a specific rule need only allege that the matter poses a significant safety problem."[223] According to Petitioners, the Board has the authority to look at what Entergy is using to "fill[] in the blanks," apart from what is explicitly defined in the Alternate PTS Rule, "and decide if that represents a bona fide, valid approach."[224] Moreover, Petitioners argue that, just as the Staff has the

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[218] Reply at 6.

[219] Tr. at 20–21, 47.

[220] Reply at 6 (citing Gundersen Declaration ¶ 27).

[221] Id. at 4.

[222] Id.

[223] Id. (citing 10 C.F.R. § 50.57(a)(3); Duke Power Co. (Catawba Nuclear Station, Units 1 & 2), LBP-82-116, 16 NRC 1937, 1946 (1982)).

[224] Tr. at 67.

authority to reject the Palisades LAR, "the Board similarly has the authority to find that an application is not complete."[225]

Petitioners also attempt to clarify their earlier statements about the 20% error band in fluence calculations, asserting more specifically that the LAR omits a study of fluence variability: "Gundersen stated . . . that there is a need for consistency in comparing the 20% error band among the sister plants and that under 10 C.F.R. §50.61a, Entergy has not made that showing."[226]

We conclude that Basis 2 is inadmissible because it conflicts with Section 50.61a(f)(6)(i) regarding the use of surveillance data in the consistency check. The purpose of the consistency check—the only portion of the Alternate PTS Rule that may require use of sister plant data—is to check the basic operation of the embrittlement model with surveillance data. The consistency check seeks to compare, for a specific material type, the model's projected embrittlement with the actual embrittlement values at the same fluence provided by material samples.[227] The Alternate PTS Rule clearly states that surveillance data must be used in the consistency check when it is (A) "a heat-specific match for one or more of the materials for which RT_{MAX-X} is being calculated," and (B) "three or more different neutron fluences exist for a specific material."[228] Thus, the use of a material sample in the consistency check is not dependent on its location inside an RPV, or which RPV it comes from.[229] If we were to limit the material samples that may be used in the consistency check to those from a particular location from a particular RPV, we

[225] Id. at 131–32.

[226] Reply at 8; Tr. at 31.

[227] 10 C.F.R. § 50.61a(f)(6)(i)(B); 75 Fed. Reg. at 16.

[228] 10 C.F.R. § 50.61a(f)(6)(i)(A), (B).

[229] Id. § 50.61a(a)(10).

would be adding a new requirement to 10 C.F.R. § 50.61a(f)(6)(i), which is prohibited by 10 C.F.R. § 2.335.

Petitioners argue that because the sister plants are operated differently from the Palisades reactor, there is significant spatial variability in the flux and fluence between plants, making a comparison between plants impossible.[230] As support, the Gundersen Declaration points to multiple sources of data that indicate that neutron flux can vary dramatically across a reactor vessel.[231] We recognize that the neutron flux hitting a material will be different at different parts of the reactor. Any variation in flux, however, is captured in the material's fluence measurement, because fluence is the integral of flux over time. Under Section 50.61a(f)(6)(i), when the fluence of a material sample is known it must be used in the consistency check if it is of the appropriate chemical composition. The regulation's consistency check does not rely on information that is unique to a particular RPV, but instead on the chemical properties and fluence of the material samples.[232] From the standpoint of the consistency check, a material sample of the same fluence and material type is no different whether obtained from the Palisades RPV or a sister plant RPV.

We also conclude that Basis 3 is inadmissible, although in this instance the problem is the lack of support for Petitioners' argument rather than a conflict with Section 50.61a(f)(6)(i). Petitioners argue that the use of sister plant surveillance data in combination with Palisades' data violates a "binding" 20% error limit.[233] This portion of the contention is not well-explained, but the Board examined the issue in an attempt to understand Petitioners' concerns.[234] The

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[230] Amended Petition at 18; Gundersen Declaration ¶¶ 34–38.

[231] Gundersen Declaration ¶¶ 34, 35; id. ¶ 37 (citing Palisades 1990 Fluence Analysis § 4.3).

[232] See 10 C.F.R. § 50.61a, equations 5–7.

[233] Amended Petition at 18.

[234] A licensing board "may appropriately view Petitioners' support for its contention in a light that is favorable to the Petitioner," although it may not do so by ignoring other admissibility

Gundersen Declaration, in citing the alleged 20% limit, references a Palisades Fluence Evaluation Report, which in turn references Regulatory Guide 1.190, pertaining to how fluence is modelled within a single reactor.[235] Regulatory Guide 1.190 requires that a certain portion of all projections derived from a fluence model fall within 20% of empirical measurements, if these calculations are to be used as inputs into embrittlement determinations.[236] Therefore, the limit discussed by Mr. Gundersen pertains to projected fluence values for an RPV, and does not pertain to comparisons of the Palisades embrittlement model with measured fluence and embrittlement values coming from either Palisades or sister plant material samples. We therefore cannot admit Basis 3.

When the Commission has determined that compliance with a regulation is sufficient to provide for reasonable assurance of public health and safety, a licensing board cannot impose requirements that exceed those in the regulation.[237] Here, given that the Commission has made such a determination regarding Section 50.61a, the Board may only review the LAR to decide if it meets the rule's requirements; it may not impose additional requirements that a petitioner believes would better protect public health and safety. Bases 2 and 3 fail to show that Entergy's consistency check violated Section 50.61a, and thus they do not support an admissible contention.

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requirements. Ariz. Pub. Serv. Co. (Palo Verde Nuclear Generating Station, Units 1, 2, & 3), CLI-91-12, 34 NRC 149, 155 (1991); see also Houston Lighting & Power Co. (S. Texas Project, Units 1 & 2), ALAB-549, 9 NRC 644 (1979) (indicating reluctance to deny intervention petition on basis of skill in pleading).

[235] Gundersen Declaration ¶ 30 (citing Regulatory Guide 1.190).

[236] See Regulatory Guide 1.190 at 3, 31; see also Tr. at 54–55.

[237] 75 Fed. Reg. at 22; Fermi 3, LBP-09-16, 70 NRC at 255. Petitioners are also incorrect in implying that the NRC Staff has the authority to deviate from the agency's regulations. Tr. at 131–32. All agencies must adhere to their own regulations. Frizelle v. Slater, 111 F.3d 172, 177 (D.C. Cir. 1997).

E. Additional Arguments Raised by Petitioners

In addition to the contention explicitly put forward by Petitioners, the Amended Petition raises other potential challenges to the LAR. These are briefly addressed below.

1. No Significant Hazards Consideration Determination[238]

The Amended Petition contends that the Staff was incorrect in concluding that the LAR "involves no significant hazards consideration."[239] Petitioners instead argue that "there is a consequential possibility that significant hazards associated with implementation of the alternative calculation method under 10 C.F.R. § 50.61a may occur, in the form of a material underestimate of the prospects of a severe PTS incident which could lead to a LOCA [loss-of-coolant accident] involving the Palisades RPV."[240] Petitioners therefore contend that "the standards of 10 CFR § 50.92 have not been satisfied," and the Staff should not have concluded that the LAR involves a no significant hazards consideration.[241]

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[238] When a licensee submits its application for a license amendment to the NRC, it must provide the agency "its analysis about the issue of no significant hazards consideration using the standards in [10 C.F.R.] § 50.92." 10 C.F.R. § 50.91(a)(1). A final "no significant hazards consideration" determination allows the Commission to issue the challenged license amendment before the petitioner's request for a hearing is adjudicated. Id. § 50.91(a)(4) ("[T]he amendment will be effective on issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in § 2.309 of this chapter has filed a request for a hearing."). But such a determination does not either prevent the adjudication from proceeding or restrict the licensing board's substantive determination on public health and safety issues. Long Island Lighting Co. (Shoreham Nuclear Power Station, Unit 1), LBP-91-7, 33 NRC 179, 183 (1991) ("A determination of no significant hazards consideration is not a substantive determination of public health and safety issues for the hearing on the proposed amendment.").

[239] Amended Petition at 9 (quoting 79 Fed. Reg. at 58,815) (internal quotation marks omitted).

[240] Id.

[241] Id.

This argument does not lead to an admissible contention. A "no significant hazards consideration" determination is a procedural decision barred from litigation pursuant to 10 C.F.R. § 50.58(b)(6) and licensing board precedent.[242]

2. Operation as a Test or Experiment

Petitioners repeatedly assert in the Amended Petition that operation of Palisades without pulling more material samples from the RPV "means that Entergy may be operating Palisades as a *test* according to 10 C.F.R. § 50.59."[243] Mr. Gundersen states that, given Palisades' alleged status as "one of the most embrittled reactors in the United States,"[244] "its continued operation as an embrittlement experiment, likely in violation of 10 CFR 50.59," will render Palisades "the symbol of a regulator-endorsed national test attempting to determine how long a damaged vessel can continue to operate."[245] Mr. Gundersen proposes that the LAR, according to 10 C.F.R. § 50.59, should "trigger the requirement for additional public scrutiny in the form of a public licensing process."[246]

This argument does not lead to an admissible contention. Section 50.59 defines what activities the licensee may pursue without submitting a license amendment request, including certain "tests or experiments."[247] Since Entergy is seeking a license amendment for use of the Alternate PTS Rule, Petitioners' argument is misplaced.

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[242] Entergy Nuclear Vt. Yankee, LLC et al. (Vt. Yankee Nuclear Power Station), LBP-04-28, 60 NRC 548, 560–61 (2004).

[243] Amended Petition at 12 (quoting Gundersen Declaration ¶ 8) (internal quotation marks omitted).

[244] Gundersen Declaration ¶ 50.

[245] Id. ¶ 16.

[246] Id. ¶ 50.

[247] 10 C.F.R. § 50.59(c)(1).

3. Chemical Composition of Sister Plant Material Samples

Petitioners contend in their reply that "Gundersen [in his declaration] has attested to the lack of proof that the metals from the various RPVs match."[248] Petitioners appear to argue that the sister plant material samples fail to support the consistency check because they are of different chemical compositions than the materials found in the Palisades RPV.[249] Petitioners alleged at oral argument that, although this statement was made in their reply, it reflects a position taken by Mr. Gundersen with the filing of the original petition to intervene.[250]

Entergy disputed at oral argument whether this argument was actually made by Mr. Gundersen in his declaration.[251] The Staff separately responded that the equations underlying the Alternate PTS Rule account for "differences between the impurities" among material samples of the same heat.[252]

Under Section 50.61a, material samples that are to be used for the consistency check must be of the same "heat."[253] As noted above, the term "heat" or "heat-specific match" is not defined in the Alternate PTS Rule; however, it is clear that the essence of the requirement is that the materials be of the same composition.[254]

Our review of the Gundersen Declaration indicates that Mr. Gundersen did not raise the argument that the sister plant material samples are of different chemical composition compared to Palisades' samples. Instead, Mr. Gundersen admits that the sister plant and Palisades

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[248] Reply at 5.

[249] Id.

[250] Tr. at 47.

[251] Id. at 87.

[252] Id. at 118.

[253] 10 C.F.R. § 50.61a(f)(6)(i)(A).

[254] Supra note 69; see 75 Fed. Reg. at 16.

samples are similar.[255] Therefore, Petitioners inappropriately raised this argument in their reply.[256]

In addition, their argument is without support and contradicts the statement of their expert. The Board's review of the Palisades Alternate PTS Rule Evaluation shows no reason to doubt that the sister plant material samples are the same "heat" or composition compared to the materials in the Palisades RPV. Although the sister plant material samples do have slightly different amounts of copper, nickel, phosphorous, and manganese than the materials in the Palisades RPV,[257] these differences are accounted for in the consistency check.[258]

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[255] Gundersen Declaration ¶ 27 ("[I]t is true that the material used to weld the reactor plates together to create the reactor vessel is similar among the four plants . . . .").

[256] La. Energy Servs., LP (Nat'l Enrichment Facility), CLI-04-25, 60 NRC 223, 225 (2004).

[257] Palisades Alternate PTS Rule Evaluation, tbls. 6-2 to 6-3.

[258] See 10 C.F.R. § 50.61a(f)(6), equations 5–7 (accounting for differing amounts of copper, nickel, phosphorous, and manganese between material samples for the consistency check); Tr. at 118–19.

VI. Conclusion

Although Petitioners have demonstrated standing to intervene, they have not put forward an admissible contention. Therefore, their petition to intervene and request for a hearing is denied. Petitioners may appeal this decision to the Commission pursuant to 10 C.F.R. § 2.311(c), within twenty-five days of service of this Order.

It is so ORDERED.

THE ATOMIC SAFETY
AND LICENSING BOARD

/RA/

Ronald M. Spritzer, Chair
ADMINISTRATIVE JUDGE

/RA/

Dr. Gary S. Arnold
ADMINISTRATIVE JUDGE

/RA/

Dr. Thomas J. Hirons
ADMINISTRATIVE JUDGE

Rockville, Maryland
May 8, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of ENTERGY NUCLEAR OPERATIONS, INC. (Entergy) (Palisades Nuclear Plant)	)))))))	Docket No. 50-255-LA

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing **LICENSING BOARD MEMORANDUM AND ORDER LBP-15-17 (Ruling on Petition to Intervene and Request for a Hearing)** have been served upon the following persons by Electronic Information Exchange.

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[Original signed by Brian Newell]
Office of the Secretary of the Commission

Dated at Rockville, Maryland
this 8th day of May, 2015