UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
Before the Commission

In the Matter of:	)	
		Docket No. 50-255
Entergy Nuclear Operations, Inc.	)	
(Palisades Nuclear Plant)		June 2, 2015
	)	
Operating License Amendment Request		
	)	

*   *   *   *   *

<u>INTERVENORS' 10 C.F.R. § 2.311(c) NOTICE OF APPEAL OF ATOMIC SAFETY AND LICENSING BOARD'S DENIAL OF PETITION TO INTERVENE AND REQUEST FOR A HEARING ON ENTERGY LICENSE AMENDMENT REQUEST FOR AUTHORIZATION TO IMPLEMENT 10 CFR § 50.61a AND BRIEF IN SUPPORT</u>

Terry J. Lodge (OH #0029271)
316 N. Michigan St., Ste. 520
Toledo, OH 43604-5627
(419) 255-7552
Fax (419) 255-7552
tjlodge50@yahoo.com

Counsel for Petitioners

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PETITIONERS' 10 C.F.R. § 2.311(c) NOTICE OF APPEAL OF ATOMIC SAFETY AND LICENSING BOARD'S DENIAL OF 'PETITION TO INTERVENE AND REQUEST FOR A HEARING ON ENTERGY LICENSE AMENDMENT REQUEST FOR AUTHORIZATION TO IMPLEMENT 10 C.F.R. § 50.61a'

Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future – Shoreline Chapter (Shoreline), and the Nuclear Energy Information Service (NEIS) (collectively "Petitioners"), by and through counsel, pursuant to 10 C.F.R. § 2.311(c), hereby give notice of their appeal to the U.S. Nuclear Regulatory Commission ("Commission") for review of the Atomic Safety and Licensing Board's ("ASLB") "Memorandum and Order (Ruling on Petition to Intervene and Request for a Hearing", LBP–15-17 (May 8, 2015) wherein the ASLB denied Petitioners' "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.'"

According to 10 C.F.R. § 2.311(c), "An order denying a petition to intervene, and/or request for hearing . . . is appealable by the requestor/petitioner on the question as to whether the request and/or petition should have been granted." Petitioners intend to urge on appeal that their petition to intervene and request for a hearing should have been granted.

/s/ Terry J. Lodge

Terry J. Lodge (OH #0029271)
316 N. Michigan St., Ste. 520
Toledo, OH 43604-5627
(419) 255-7552
Fax (419) 255-7552
Tjlodge50@yahoo.com
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BRIEF IN SUPPORT OF PETITIONERS' 10 C.F.R. § 2.311(c) APPEAL OF ATOMIC SAFETY AND LICENSING BOARD'S DENIAL OF 'PETITION TO INTERVENE AND REQUEST FOR A HEARING ON ENTERGY LICENSE AMENDMENT REQUEST FOR AUTHORIZATION TO IMPLEMENT 10 C.F.R. § 50.61a'

I. Introduction

This proceeding concerns Entergy Nuclear Operations, Inc.'s ("Entergy's") request to amend the operating license for the Palisades nuclear plant ("Palisades"). Palisades is a single pressurized water reactor ("PWR") facility located on the eastern shore of Lake Michigan, five miles south of South Haven, Michigan. 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. That alternate method is set forth in an agency regulation, "Alternate fracture toughness requirements for protection against pressurized thermal shock events." In an operating nuclear power plant, the reactor vessel is continuously exposed to neutrons from fission reactions occurring inside the vessel. Over time, this neutron radiation embrittles the RPV walls, making them less able to resist fracturing, i.e., "fracture toughness" decreases. 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. Of significant concern is a pressurized thermal shock, or "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."[1] 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."[2]

On September 30, 2014, the NRC Staff (the Staff) published notice of Entergy's LAR, and concluded that the LAR presents "no significant hazards consideration" under 10 C.F.R. § 50.92(c). In response to the LAR notice, Petitioners filed the instant petition to intervene and request for a hearing.[3]

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

[2]*Id.* at xix; *see also* "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Final Rule," 75 Fed. Reg. 13, 14 (Jan. 4, 2010). 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).

[3]"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'"

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.

Petitioners' hearing request was referred to an Atomic Safety and Licensing Board for consideration. Both Entergy and the NRC Staff filed answers opposing the Amended Petition, to which Petitioners filed a reply. On March 25, 2015, the Board heard oral argument on standing and contention admissibility, and on May 8, 2015, the ASLB issued its "Memorandum and Order (Ruling on Petition to Intervene and Request for a Hearing"), LBP–15-17 wherein the ASLB denied Petitioners' Amended Petition to Intervene and for a Public Adjudication Hearing.

II. Factual and Procedural Background

A. The 1985 PTS Rule And 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.[4] The program to monitor PWR RPVs is described in 10 C.F.R. Part 50, Appendix H,

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(December 8, 2014) (hereinafter "Amended Petition").

[4] *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).

and is titled "Reactor Vessel Material Surveillance Program Requirements" (Surveillance Program). 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."[5] The Surveillance Program relies on physical material samples, also known as specimens, capsules, or coupons, "which are withdrawn periodically from the reactor vessel."[6] The NRC must pre-approve the schedule for removing material samples from the reactor vessel.[7]

The actual screening limits required by Appendix H's Surveillance Program for monitoring reactor pressure vessels ("RPVs") for fracture toughness are established in 10 C.F.R. § 50.61, entitled "Fracture toughness requirements for protection against pressurized thermal shock events." Section 50.61 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.[8]

NRC regulations represent steel fracture toughness as a temperature value, known as "reference temperature." The NRC Staff says, "[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

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[5]10 C.F.R. Part 50, App. H(I).

[6]*Id.* 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.* Part 50, App. H(III)(B)(2).

[7]*Id.* Part 50, App. H(III)(B)(3).

[8]*See id*. § 50.61(c)(2)(i).

a low reference temperature."[9] 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. But at low temperatures, steel is naturally brittle, and even unirradiated steel can potentially suffer brittle failure.[10] 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."[11] As described by Staff guidance documents, this transition point depends primarily on two factors material composition and cumulative irradiation by high-energy neutrons.[12] As steel is exposed to more high-energy neutrons (i.e., its fluence increases),[13] RT_{NDT} increases concurrently.[14] Thus, as fluence increases,

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[9]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").

[10]*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.

[11]*Id*. at xxxiv. "NDT" stands for Nil-Ductility Transition. *Id.* at xxxi.

[12]*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).

[13]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(cubed)-second, typically expressed as neutrons/centimeter (squared)-second. *See* Samuel Glasstone and Alexander Sesonske, Nuclear Reactor Engineering § 2.118 (Van Nostrand Reinhold Co. 1967).

[14]*See* Alternate PTS Rule Technical Basis Report § 2.4.1 (discussing the reference temperature approach to characterizing fracture toughness in ferritic materials).

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, which are 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 event.[15] The PTS "screening criterion" is 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld materials."[16]

If the RT_{NDT} values projected at specific areas of the RPV for the end of life of the plant, known as RT_{PTS},[17] 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.[18] 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

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[15]*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.

[16]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").

[17]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").

[18]10 C.F.R. § 50.61(b)(3)–(5).

to "reduce the potential for failure of the reactor vessel due to PTS events."[19]

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

While no reactor is expected to exceed the current screening criteria established in 10 C.F.R. § 50.61 during its 40 year operating license, the Staff has noted that Palisades in particular is one of the first plants likely to exceed them, as Palisades' RPV is "constructed from some of the most irradiation-sensitive materials in commercial reactor service today."[20] This concern, as well as significant advancements in failure analysis and materials knowledge, prompted the NRC to reexamine the § 50.61 approach for projecting fracture toughness and the screening criteria.[21] 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."[22] The report concluded that through-wall cracks were much harder to create in RPVs than initially thought, and occurred in fewer circumstances.[23] 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

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[19] *Id.* § 50.61(b)(6).

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

[21] *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.

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

[23] *See id.* at xx–xxiii.

RPV.[24] 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.[25]

On October 3, 2007, the Staff published a notice of proposed rulemaking.[26] 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," and that "[t]his finding indicates that the screening criteria in 10 CFR 50.61 are unnecessarily conservative."[27]

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.[28] 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.[29] The Alternate Screening Criteria consist of eighteen different reference temperature limits that depend on RPV

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

[25]*Id.* at xxvii.

[26]72 Fed. Reg. 56,275.

[27]*Id.* at 56,276.

[28]However, 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 Rule's screening criteria. Id. § 50.61a(d)(2)–(7).

[29]75 Fed. Reg. at 18.

wall thickness and the part of the RPV under consideration.[30] The Alternate PTS Rule also changes how licensees derive projected reference temperatures for the components of their RPVs.[31] 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."[32] Section 50.61, by contrast, continuously integrates surveillance data into future embrittlement projections.[33] 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."[34] 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."[35] Furthermore, the final rulemaking stated that "[t]he final rule will not significantly

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[30] 10 C.F.R. § 50.61a(g) tbl. 1.

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

[32] *Id.*

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

[34] 75 Fed. Reg. at 18.

[35] *Id.* at 22.

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."[36]

C. Invocation Of The Alternate PTS Rule

To take advantage of the Alternate PTS Rule, a licensee must request approval from the NRC 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.[37] 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 to experimental measurements and with quantified uncertainties and possible biases."[38] From that point, the licensee must establish $RT_{NDT(U)}$ for various key points along the RPV.[39] 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

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[36] *Id.*

[37] 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.

[38] *Id.* § 50.61a(f).

[39] *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).

$RT_{MAX\text{-}X}$.[40] The embrittlement model allows for calculations of $RT_{MAX\text{-}X}$ across the RPV using probabilistic analyses, without having to rely on measured data.[41] The $RT_{MAX\text{-}X}$ values are compared to the Alternate Screening Criteria to determine whether the RPV is safe to operate.[42] Importantly, as calculations of $RT_{MAX\text{-}X}$ 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.[43] 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, App. H, surveillance programs."[44] The purpose of the check is to "determine if the surveillance data show a significantly different trend than the embrittlement model predicts."[45] The check includes three statistical analyses that compare the model's inputs, fluence and material properties, with the model's output, reference temperature.[46]

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[40] *Id.* § 50.61a(f)(1)–(3). "$RT_{MAX\text{-}X}$ 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)

[41] *See supra* note 34.

[42] *See* 10 C.F.R. § 50.61a(c)(3).

[43] *Id.* § 50.61a(f)(6)(i).

[44] 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).

[45] 10 C.F.R. § 50.61a(f)(6)(i)(B).

[46] 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

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."[47]

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 for the deviations from the Director of the Office of Nuclear Reactor Regulation.[48]

D. Petitioners' Objections To Entergy License Amendment Request (LAR) Invoking Alternate PTS Rule

On September 30, 2014, notice was published in the Federal Register[49] of Entergy's intentions of seeking amendment of the operating license of Palisades Nuclear Plant to allow implementation of an alternative method of calculation of the degree of embrittlement of the Palisades nuclear reactor pressure vessel. The 10 C.F.R. § 50.61 screening criteria, to which Palisades supposedly adhered, define a limiting level of embrittlement beyond which plant operation cannot continue without further evaluation. The switch to the use of 10 CFR § 50.61a will change how fracture toughness of the reactor vessel is determined, moving from an analytical to a probabilistic risk assessment method. Entergy's proposed "no significant hazards" determination, required by 10 C.F.R. § 50.91(a), concluded that the proposed change will not involve a significant increase in the probability or consequences of an accident previously

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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).

[47] 10 C.F.R. § 50.61a(f)(6)(i)(B).

[48] *Id*. § 50.61a(f)(6)(vi).

[49] 79 Fed. Reg. 58812 (September 30, 2014)

evaluated.[50] Entergy further concluded that the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.[51] The utility maintained, also, that the proposed change would not involve a significant reduction in a margin of safety.[52] In light of Entergy's analysis, the NRC Staff concluded that "the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration."[53]

When the Palisades RPV was brand new, its reference temperature-nil ductility transition (RT-ndt) was at 40 degrees F. By the early 1980s, NRC had weakened Palisades' screening criteria - and the rest of the U.S. pressurized water reactors' - to 200 degrees F, which is closer to the operating temperature of Palisades, which is around 550 degrees F. Thus if the Emergency Core Cooling System ("ECCS") pumps too-cold water into the 550 degrees F reactor pressure vessel and cools it too quickly down to 200 degrees F (or, later, 270 or 300 degrees), there instantaneously arises a serious potential for a fracture of the RPV, which would be a very significant reactor accident. When the PWR safety system repressurizes the RPV, the metal can't take it any more, and fractures. It breaks, either by major cracking or actual fragmentation, presumably at the point of a flaw in the RPV.

As noted, 200 degrees F was merely an early retreat from regulation. The criteria were later relaxed to 270 degrees F for axial/vertical welds, and to 300 degrees F for welds of a

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[50]*Id*. at 58815.

[51]*Id.*

[52]*Id.*

[53]*Id.*

circumferential/horizontal orientation. And through it all, Palisades and/or the NRC have projected, again and again that the new PTS screening criteria would be exceeded by a predicted future date. These dates have been 1995; 1999; September 2001; 2004; 2007; 2014; April 2017; and August 2017. On or near those dates, Palisades or the NRC has said, the allowable boundary beyond which lies the risk of disaster will be crossed. Each time, though, the date of heightened vulnerability to this type of disaster has routinely slipped back further into the future.

In the many years since the early indicators of embrittlement in its first operational decade, Palisades has gained notoriety as one of the nation's most-embrittled reactors. In its May 19, 1995 NRC Generic Letter 1992-001, Supplement 1,[54] the NRC Staff permitted Palisades to operate until late 1999, observing that it had "reviewed the other PWR vessels and, based upon currently available information, believes that the Palisades vessel will reach the PTS screening criteria by late 1999, *before any other PWR*." (Emphasis added). *Id.*

Petitioners' objections to the ASLB relied in large part on the expert opinion of nuclear engineer Arnold Gundersen (*see* "Declaration of Arnold Gundersen," hereinafter "Gundersen Declaration") that the analysis provided to the NRC by Entergy is inadequate and relies upon unsupported assumptions which warrant a hearing as to whether Entergy should be allowed to switch over to 10 C.F.R. § 50.61a. Petitioners urged the possibility exists that significant hazards associated with implementation of the alternative calculation method under 10 C.F.R. § 50.61a may occur, caused by materially-underestimated prospects of a severe loss-of-coolant accident (LOCA) involving the reactor.

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[54]ADAMS No. ML031070449.

Arnold Gundersen stated that “Almost half of the initial capsules [coupon samples] installed 43 years ago still remain inside the embrittled nuclear reactor” and that if the NRC allows Entergy to postpone the next Palisades coupon sampling until 2019, “then no accurate current assessment of Palisades’ severe embrittlement condition exists.” Gundersen Declaration p. 8, ¶ 21. Gundersen opined that § 50.61 is analytical in nature, while § 50.61a authorizes probabilistic risk assessment, and that the discretionary availability of § 50.61a under the circumstances cannot be used as a substitute for scientific investigation. *Id.* at p. 9, ¶ 24.3. Gundersen observed (*id.* at p. 3, ¶ 8) that “Continued operation of the Palisades nuclear power plant without analyzing the coupon designated to be sampled more than seven years ago means that Entergy may be operating Palisades as a *test* according to 10 C.F.R. § 50.59.” (Emphasis in original).

Petitioners’ expert further alleged that the underlying data from other supposedly comparative nuclear plants assessing ductility of their RPVs is not legitimate: “The NRC has allowed Palisades to compare itself to reactors of disparate designs from other vendors, built in different years and operating at diverse power levels.” Gundersen Declaration at ¶ 24.2. These plants, which he says “thus far have not exhibited significant signs of reactor metal embrittle-ment,” are poor comparables because:

> . . . the dramatically different nuclear core design and operational power characteristics make an accurate comparison impossible. The difference between the Westinghouse nuclear cores and the Combustion Engineering nuclear core impacts the neutron flux on each reactor vessel, thus making an accurate comparison of neutron bombardment and embrittlement impossible.

Id. at p. 10, ¶ 27.

The core objection raised by Petitioners’ filing is that the 10 C.F.R. § 50.61a alternative to § 50.61 allows Entergy to substitute various estimates of the status of the RPV for actual data investigation and analysis. Those § 50.61a projections are attained, among other means, by averaging data on reactor vessels from other nuclear power plants, to arrive at a projection of the current status of the Palisades RPV. Entergy's recourse to the alternate approach, accompanied as it is by deliberate non-testing of metal coupons from the RPV for 16 years (2003-2019) can be understood only if one assumes that Entergy does not want to know what physical testing might attain by way of useful data about the true state of affairs within the Palisades RPV.

As Petitioners' expert, Arnold Gundersen objected to the specific comparable nuclear reactor vessels cited by Entergy to comply with § 50.61a, pointing out that "The NRC has allowed Palisades to compare itself to reactors of disparate designs from other vendors, built in different years and operating at diverse power levels." Gundersen Declaration at ¶ 24.2. These plants, which he said "thus far have not exhibited significant signs of reactor metal embrittlement," are poor comparables because:

> . . . the dramatically different nuclear core design and operational power characteristics make an accurate comparison impossible. The difference between the Westinghouse nuclear cores and the Combustion Engineering nuclear core impacts the neutron flux on each reactor vessel, thus making an accurate comparison of neutron bombardment and embrittlement impossible.

Id. at p. 10, ¶ 27.

A good example of a false comparison is found in Structural Integrity Associates, Inc.'s Report No. 0901132.401, Revision 0, "Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis," ADAMS No. ML110060693. This document was part of the technical basis for the PTS safety risk regulatory rollback of PTS screening criteria, from January 2014 to April 2017 at Limiting Beltline Weld W5214. "Similar Sister Plant" proxies were used which involved the inappropriate averaging of 11 sample surveillance capsules/coupons from very dissimilar RPVs. Ssuch false comparisons, Gundersen says, "significantly dilute Palisades' embrittlement calculations." *Id.* at p. 11, ¶ 28. He adds: "This rogue comparative data is not sound scientific methodology and clearly places the operations of the Palisades NPP in the experimental test venue, possibly as delineated in 10 CFR 50.59." *Id.* at p. 11, ¶ 29.

The most serious analytical problem in using sister plants data "is the extraordinary difficulty comparing data from four separate plants while still maintaining one standard deviation (1σ) or 20% between all the data. According to the *Palisades Reactor Pressure Vessel Fluence Evaluation*, one standard deviation is required, however there has never been a discussion of how this was achieved between the four sister units." Gundersen Declaration at p. 11, ¶ 30. While "[a] 1σ analysis appears to be binding within the Palisades data, . . . the NRC lowers the bar when comparing data from similar sister plants that are included in Entergy's analysis of the Palisades reactor vessel without requiring the same 1σ variance with Palisades." *Id*. at p. 12, ¶ 32. Gundersen added: "There can be no assurance that the 20% error band at Palisades encompasses the 20% error band at the Robinson or Indian Point plants. To compare this different data without assurance that the 1σ variance from each plant overlaps the other plants lacks scientific validity." *Id.* at p. 12, ¶ 33.

Gundersen further found that there is "extraordinary variability between the neutron flux across the nuclear core in this Combustion Engineering reactor" because of a "flux variation of as much as 300% between the 45-degree segment and the 75-degree segment," calling it "mathematically implausible that a 20% deviation is possible when the neutron flux itself varies by 300%." *Id*. at p. 12, ¶ 34. In sum, he noted that:

> The Westinghouse Analysis delineates that a 20% variation is mandatory, yet the effective fluence variability can be as high as 300%, therefore, *the analytical data does not support relicensure without destructive testing and complete embrittlement analysis of additional capsule samples*.

Id. at p. 16, ¶ 39.

III. Argument

A. The ASLB Erroneously Found The Decision Allowing Entergy To Invoke 10 C.F.R. § 50.61a To Be Nondiscretionary

The Atomic Safety and Licensing Board generally denied the Petition, holding that:

> 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.

(Emphasis supplied). LBP-15-17 at 29.

The Licensing Board's reasoning is flawed; it involves two distinct considerations. Even assuming *arguendo* that the NRC can promulgate an alternative regulation that is weaker than the other, and afford a choice of laws to nuclear utility operators, that position says nothing about the discretionary nature of the NRC Director of Nuclear Reactor Regulation over whether to allow a particular applicant to invoke 10 C.F.R. § 50.61a. The ASLB ruled, in essence, that if the paperwork is properly completed, the substantive issue - whether to allow Entergy to move to 10 C.F.R. § 50.61a - is essentially irrelevant, is to be automatically allowed, and that the NRC Staff's regulatory hand must be stayed. This dogmatic stance is apparent in several ASLB statements. For example, the ASLB adopted Entergy's argument that "a contention asserting that different analysis or technique should be utilized is inadmissible because it indirectly attacks the Commission's regulations." LBP-15-17 at 33. Petitioners were advocating, not for usage of a different technique to be used, but that that the Director of NRR should have discretionarily considered whether a superior "reasonable assurance" of protection of public health and safety would be derived from rejecting Entergy's request to invoke § 50.61a.

This is because 10 C.F.R. § 50.61a clearly contemplates a discretionary determination by the Director of NRR. *See*, for example, § 50.61a(c)(1) (RT_{MAX-X} values assessment "must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation"); § 50.61a(c)(2) ("Each licensee shall perform an examination and an assessment of flaws in the reactor vessel beltline as required by paragraph (e) of this section" - and (e) requires disclosure of tests performed but, again, detailed explanation of the methodology underlying NDE uncertainties assumptions,[55] and adjustments must be disclosed. This is merely a recognition that even objective data, once interpreted, may be examined to ascertain the objectivity or inappropriate bias which may have occurred in the means of analysis which have been applied to it. Where there is discretion vested in the regulator, differences of opinion, interpretation, and expert analysis are legitimate bases for challenging the decision because the decision is potentially arrived at in an adversarial manner.

This principle is also obvious in § 50.61a(f)(7), which requires that "The licensee shall report any information that significantly influences the RT_{MAX-X} value to the Director in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section." The requirement clearly introduces subjective judgment and selection among different conditions or findings into the decision of what data is to be provided to the Director of NRR.

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[55] § 50.61a says in part: "The methodology to account for NDE-related uncertainties must be based on statistical data from the qualification tests and any other tests that measure the difference between the actual flaw size and the NDE [no-destructive examination] detected flaw size. Licensees who adjust their test data to account for NDE-related uncertainties to verify conformance with the values in Tables 2 and 3 shall prepare and submit the methodology used to estimate the NDE uncertainty, the statistical data used to adjust the test data and an explanation of how the data was analyzed for review and approval by the Director in accordance with paragraphs (c)(2) and (d)(2) of this section."

Hence for Petitioners to provide their expert's critique of the means by which the § 50.61a investigation was conducted, and the weaknesses or biases in the underlying data, assumptions and manipulations of information cannot be construed as a frontal assault on the regulatory citadel, but must instead be seen, for purposes of the admissibility determination, as an exposition of the flaws caused by straying away from knowable science. Petitioners' critique was not answered by any experts on behalf of the NRC Staff or Entergy. Petitioners articulated challenges to the proposed exercise of discretion by the Director of Nuclear Reactor Regulation and should be accorded a hearing to provide more evidence.

The Commission should take note that the agency regulations contain a "pressurized thermal shock regulatory relief valve" for situations where a nuclear utility cannot meet even the flaccid threshold of 10 C.F.R. § 50.61a, by means of which the Director of NRR may allow an embrittled reactor to operate beyond the PTS screening criteria. *See* slide show, "Technical Brief on Regulatory Guidance on the Alternative PTS Rule (10 C.F.R. § 50.61a)," Official Transcript of Proceedings, ADAMS No. ML14321A542, at p. 242/268 of .pdf:

> Use of 10 CFR 50.61a PTS screening criteria requires submittal for review and approval by Director, NRR.
>
> For plants that do not satisfy PTS Screening Criteria, plant-specific PTS assessment is required.
>
> Must be submitted for review and approval by Director, NRR.
>
> *Guidance is not provided for this case.*
>
> Subsequent requirements (*i.e.*, after submittal) are defined in paragraph (d) of 10 CFR 50.61a. (Emphasis supplied).

B. 'Reasonable Assurance' Cannot Apply Alike To Two Regulations Addressing The Same Subject When One Is Deemed To Be Weaker Than The Other

When the ASLB referred to the 10 C.F.R. § 50.61 requirements as "more demanding" than the "Alternate Fracture Toughness Requirements," the Board agreed that the "evident purpose" of 10 C.F.R. § 50.61a is to weaken the regulatory rigor over nuclear utilities with serious RPV ductility problems. Petitioners suggest that substitution of a stronger standard which officially provides "reasonable assurance" of public protection with an admittedly weaker one also "reasonably assured" to be protective,[56] is legally anomalous.

Section 182a of the Atomic Energy Act states that a reactor operating license must include "technical specifications" that include, *inter alia*, "the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material . . . will provide adequate protection to the health and safety of the public." 42 U.S.C. § 2232(a). The general requirement for operating licenses, 10 C.F.R. § 50.57(a)(3), requires a finding of reasonable assurance of operation without endangering the health and safety of the public.[57] *Duke Power Co.* (Catawba Nuclear Station, Units 1 & 2), LBP-82-116, 16 NRC 1937, 1946 (1982). In this proceeding, Entergy must demonstrate that it satisfies the "reasonable assurance standard" by a preponderance of the evidence. Reasonable assurance "is not susceptible to formalistic quantification or mechanistic application. Rather, whether the reasonable assurance standard is met is based upon sound technical judgment applied on a case-by-case basis." *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), LBP-07-17, 66 NRC 327, 340 (2007),

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[56]The "reasonable assurance" finding of 10 C.F.R. § 50.61a is found at 75 Fed. Reg. at 22.

[57]"(a) Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:
(1) ***;
(2) ***;
(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public. . .".

aff'd, CLI-09-07, 69 NRC 235, 263 (2009) (rejecting an argument that reasonable assurance should be quantified with 95% confidence). To consider a stronger regulation and a weaker one to be on the same footing when it comes to providing reasonable assurance is logically inconsistent, as illustrated by this very case. Palisades contains the worst-embrittled reactor pressure vessel in the United States. Posed a choice between a tougher, physical testing-based regulatory regime, or a weaker, projective method of assessing RPV ductility, owners of the worst-embrittled reactor have chosen the less-protective regulations. Because they are less protective, and given the enormous discretion vested in the Director of Nuclear Reactor Regulation to decide on a case-by-case basis what terms and conditions should be imposed under 10 C.F.R. § 50.61a, a hearing is necessary to resolve factual issues in line with regulatory expectations. The ASLB's candor shows that the alternative regulation exists merely to provide Entergy with "reasonable assurance" of being able to operate Palisades in disregard of the destructive testing obligations of 10 C.F.R. § 50.61 and in derogation of the binding requirement of reasonable assurance that the public's health and safety will be the priority for protection.

C. Variabilities In Sister Plant Data Erroneously Allowed Inappropriate Comparisons

The ASLB treated Petitioners' objections to the invalidity of sister plant data as attempts to suggest regulatory parameters which exceed the requirements of 10 C.F.R. § 50.61a. But Petitioners have previously argued that the considerable discretion accorded the Director of NRR to allow invocation of § 50.61a should be construed as lending relevance to their apples/oranges quibbling. Further, 10 C.F.R. § 50.61a(f)(6)(i) requires that "(A) The surveillance material must be a heat-specific match for one or more of the materials for which RT_{MAX-X} is being calculated." Petitioners' expert Gundersen attested to the lack of proof that the metals from the various RPVs match. This conclusion was not rebutted by any expert evidence from either the NRC Staff nor Entergy. The Licensing Board's implicit finding that the metals compared in the sister plants workup were "of the appropriate chemical composition" (LBP-15-17 at 41) was seriously challenged by Petitioners' expert witness. Nor did Entergy or the NRC Staff refute Gundersen's observation that (noted at p. 17 *infra*) that there is "extraordinary variability between the neutron flux across the nuclear core in this Combustion Engineering reactor" because of a "flux variation of as much as 300% between the 45-degree segment and the 75-degree segment," and concluding it was "mathematically implausible that a 20% deviation is possible when the neutron flux itself varies by 300%." Gundersen Declaration p. 12, ¶ 34. Perhaps § 50.61a is the culmination of decades of learning about embrittlement, but it still cannot dispense with huge variations in neutron flux in Palisades, alone. The ASLB improperly rejected this portion of Petitioners' contention.

IV. Conclusion

The threshold admissibility requirements of NRC's contention rule should not be turned into a "fortress to deny intervention." *Power Authority of the State of New York, et al.* (James FitzPatrick Nuclear Power Plant; Indian Point Nuclear Generating Unit 3), CLI-00-22, 52 NRC 266, 295 (2000). There is no requirement that the petitioners' substantive case be made at the contention stage. *Matter of Entergy Nuclear Generation Co., et al.* (Pilgrim Nuclear Power Station), 50-293-LR (ASLB Oct. 16, 2006), 2006 WL 4801142 at (NRC) 85. The Commission has explained that the requirement at § 2.309(f)(1)(v) "does not call upon the intervenor to make its case at [the contention] stage of the proceeding, but rather to indicate what facts or expert opinions, be it one fact or opinion or many, of which it is aware at that point in time which provide the basis for its contention." *Pilgrim* at 84. The admissibility requirement "generally is fulfilled when the sponsor of an otherwise acceptable contention provides a brief recitation of the factors underlying the contention or references to documents and texts that provide such reasons." *Id.*

WHEREFORE, the adverse determinations of the Atomic Safety and Licensing Board in LBP-15-17 should be reversed and the matter remanded to the ALSB for an evidentiary hearing.

Respectfully submitted,

/s/ Terry J. Lodge

Terry J. Lodge (OH #0029271)
316 N. Michigan St., Ste. 520
Toledo, OH 43604-5627
(419) 255-7552
Fax (419) 255-7552
Tjlodge50@yahoo.com
Counsel for Petitioners

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
Before the Commission

In the Matter of	)	Docket No. 50-255
Entergy Nuclear Operations, Inc. (Palisades Nuclear Plant)	)	June 2, 2015
	)	
Operating License Amendment Request	)	

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CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "PETITIONERS' 10 C.F.R. § 2.311(c) NOTICE OF APPEAL OF ATOMIC SAFETY AND LICENSING BOARD'S DENIAL OF 'PETITION TO INTERVENE AND REQUEST FOR A HEARING ON ENTERGY LICENSE AMENDMENT REQUEST FOR AUTHORIZATION TO IMPLEMENT 10 C.F.R. § 50.61a'" and the accompanying "BRIEF IN SUPPORT" were served by me upon the parties to this proceeding via the NRC's Electronic Information Exchange system this 2nd day of June, 2015.

/s/ Terry J. Lodge

Terry J. Lodge (OH #0029271)
316 N. Michigan St., Ste. 520
Toledo, OH 43604-5627
(419) 255-7552
Fax (419) 255-7552
Tjlodge50@yahoo.com
Counsel for Petitioners