NRC Staff Responses to Public Comments on DG-1299 and NUREG-1263

DG-1299, "Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule"
NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule"

· **Federal Register 80 FR 13449 (March 13, 2015)** ·

The Public Comment period ended on May 12, 2015.

I. INTRODUCTION

This document presents the NRC's responses to written public comments received on two documents:

- Draft Regulatory Guide (DG)-1299 (Revision 0 of RG 1.230), "Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule" (ADAMS Accession Number ML14056A011) and
- NUREG-2163, "Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule" (ADAMS Accession Number ML15058A677).

These comments were received in response to a *Federal Register* notice (80 FR 13449, March 13, 2015).

II. OVERVIEW OF COMMENTERS AND COMMENTS

The staff received 17 comment submissions. Table 1 presents information on these comments. In the remainder of this document each submission will be referenced by its identifier, which appears in the last column.

Table 1 – Summary of comment submissions

Commenter Name	Commenter Affiliation	Incoming ADAMS Accession No.	Identifier
Lewis Cuthbert	Alliance for Clean Environment	ML15124A023	ACE-1
Jack Stringfellow	Pressurized Water Reactor Owners Group	ML15138A093	PWROG-2
Anne Demma	Electric Power Research Institute	ML15139A026	EPRI-3
Kevin Kamps	Beyond Nuclear	ML15139A027	BN-4
Kevin Kamps	Beyond Nuclear	ML15139A028	BN-5
Kevin Kamps	Beyond Nuclear	ML15139A029	BN-6
Kevin Kamps	Beyond Nuclear	ML15139A030	BN-7
Kevin Kamps	Beyond Nuclear	ML15139A031	BN-8
Kevin Kamps	Beyond Nuclear	ML15139A032	BN-9
Kevin Kamps	Beyond Nuclear	ML15139A083	BN-10
Kevin Kamps	Beyond Nuclear	ML15139A033	BN-11
Kevin Kamps	Beyond Nuclear	ML15139A034	BN-12
Anonymous	Unknown	ML15139A035	ANON-13
Michael J. Kegan	Don't Waste Michigan	ML15139A036	DWM-14
Kevin Kamps	Beyond Nuclear	ML15139A082	BN-15
Kevin Kamps	Beyond Nuclear	ML15139A025	BN-16
Kevin Kamps	Beyond Nuclear	ML15146A057	BN-17

Some submissions are complex and present a number of different individual comments. In

those cases a sequential number is added to the end of the identifier for clarity.

Comment ACE-1

ACE strenuously objects to NRC's proposed draft regulatory guide DG-1299 and accompanying draft NUREG-2163 that would weaken nuclear reactor embrittlement standards.

NRC's is failing to require stringent independent testing for metal fatigue and microcracking in U.S. reactors continually degraded by the reactive process.

In essence, NRC's proposal would provide yet another dangerous loophole, which is just the latest in a series of NRC changes that weaken NRC's original safety standards.

Recently, news has been pouring in concerning reactor pressure vessel (RPV) flaws at nuclear reactors around the world.

This proposal appears to be another of NRC's tactics to avoid requiring regulatory compliance for embrittlement testing in U.S. reactors. NRC's proposal appears to be in response to the shocking unexpected discovery of 16,000 cracks in two Belgium reactor pressure vessels (RPVs).

Belgium's nuclear regulators urged nuclear agencies across the world to conduct the same tests that were done in Belgium to determine whether nuclear reactor microcracking may be an endemic problem for reactors. We agree with Belgium's regulators that the same testing used to identify cracks in Belgium reactors should be used worldwide, including at Limerick Nuclear Plant.

After learning that 16,000 cracks were found in just two Belgium reactors, 3-16-15, ACE requested material fatigue testing of Limerick Nuclear Plant's two reactors because NRC inspections reports have already identified material fatigue problems at Limerick. Yet, testing was never done and NRC failed to require this testing for Limerick relicensing.

Independent Reactor Embrittlement Testing Is Essential To Prevent Core Meltdowns And Catastrophic Radioactive Releases.

Embrittlement causes microcracks, invisible until cracks become large enough to be seen or until radioactive water leaking from the reactor is discovered. Leaking could cause the nuclear reactor's core to be exposed and the core's fission chain reaction would spiral out of control, initiating catastrophic nuclear meltdown.

Embrittlement of Reactor Metal Is Caused By Radiation Exposure

- *Reactor Pressure Vessel (RPV) metal walls cover the atomic core.*
- *RPV metal becomes embrittled by being continuously exposed to the radioactive neutron bombardment, resulting from the core's fission chain reaction.*
- *“Fault lines” of microscopic cracks due to metal fatigue are caused by this exposure.*
- *Undetected "fault lines" can line up creating a crack in the RPV that allows water to leak out of the vessel, exposing the core.*

- *Undetected embrittlement could even result in the reactor unexpectedly shattering like glass, initiating catastrophic nuclear meltdown.*

NRC's Draft Regulatory Guide Proposal, In Lieu Of Embrittlement Testing To Prove Safety Is Indefensible.

At Limerick, NRC's own documents reveal that the degraded condition of Limerick's reactor structures and associated equipment has been seriously impacted by the constant radioactive stresses of Limerick's 30 years of operations. Those stresses are dramatically compounded by the inherent design flaws in Limerick's GE Mark II Boiling Water Reactors, and Limerick's substandard concrete.

Our review of Exelon's requests for Limerick amendments, reliefs, and exemptions from material fatigue testing suggests that reactor embrittlement could be dangerously pervasive at Limerick.

Material Fatigue Is Caused By Stresses Put On The Reactors During Routine Operations. Other Stresses, Like Scrams, Compound The Problem.

Limerick Has Been Plagued With A Long History Of Scrams That Could Have Caused Substantial Reactor Embrittlement.

Limerick's history of SCRAM events (planned or unplanned shutdowns) illustrates that aside from neutron bombardment, a great deal of stress has been placed on the entire reactor system and supporting equipment. It is troubling that the root cause of many Limerick scrams has never been identified.

<u>Examples of Limerick's Many Scrams:</u>
NRC's 1-31-12 RAI reveals that there were 14 Limerick scrams in 2011. Records show that Exelon has a pervasive history of failing to properly analyze, determine, and correct the root cause of many of the 2011 scram events.

Examples include:
- *2-25-11 - Unit 2*
- *4-2-11 - Unit 2*
- *5-29-11 - Unit 2*
- *5-30-11 - Unit 2*
- *6-3-11 - Unit 1*

Other Examples of Limerick scrams
- *7-11-12- Unit 1*
- *7-18-12 - Unit 1 - scram and explosion*
- *7-27-12 - Unit 2*
- *8-31-12 - Unit 1*
- *3-5-14 -Unit?*
- *3-11-14 - Unit 1*
- *2-24-15 - Unit 1*

The 7-11-12 scram revealed reactor fatigue cracks.
- *In 1984, it was reported that hundreds of Limerick's safety-related welds were not properly welded by the Bechtel Power Corp. welders and that welds were not properly inspected by Bechtel and NRC inspectors.*

- *This is especially alarming due to the fact that the 7-11-12 scram was caused by the inoperability of two independent reactor channels, indicating they were subject to vibration.*
- *Fatigue cracks were observed along the weld toe due to reverse bending. Yet NRC granted Exelon "Relief Requests" for weld inspections, irrationally counting relief as compliance for relicensing. In essence, NRC is allowing elimination of a requirement to be a substitute for compliance.*

Embrittlement Issues Are Compounded By Limerick's Inherently Defective Reactors.

- *In April, 1972 a GE Mark II BWR caused a nuclear accident due to the fact that the reactor design could not hold up to the intense vibrations in the reactor created by the cooling process.*
- *All GE Mark II boiling water reactors, including Limerick's, are inherently defective.*
- *Unlike Limerick, some nuclear plants scrapped their plants or sued GE due to this reactor defect.*
- *However, on November 15, 1972 the first component of Limerick's defective GE Mark II BWR arrived on site.*
- *Supports added to Limerick's reactors to reduce its vibrations would not reduce the vibrating forces inside the reactors.*

New Understandings About Embrittlement Suggest That Specific Embrittlement Testing Is The Only Way To Accurately Determine The Safety Condition Of Limerick's Reactor Structures And Equipment.

NRC Did NOT Require Limerick-Specific Embrittlement Testing. Instead, NRC Granted Exelon's Relief Requests For Limerick, Counting Relief As Compliance For Relicensing. This Could Be Happening Elsewhere.

Testing recommended by some nuclear experts to accurately identify embrittlement and microcracking includes:

- *Ultrasonic Testing assesses embrittlement damage to the RPV and for concrete ASR damage.*
  The newest recommendation is that RPVs should have the newer ultrasonic survey which can scan a much larger area of the vessel.
- *Metal Coupons (in the RPV).*
  Coupons are metal samples that are put in the RPV and later removed for examination of degradation to track of the severity of RPV embrittlement. They are more accurate than computer modeling and extrapolations (inferred deductions) that only estimate (and can hide) actual embrittlement.
- *Monitoring "Scrams" ("startups" & "planned or unplanned shutdowns").*
  Transients or scrams cause cyclic strains contributing to reactor fatigue stress
- *Petro-graphic Analysis for Concrete Alkali-Silica Reaction (ASR) damage.*
  Visual inspection does not identify ASR, confirm ASR, or provide the current state of ASR damage (if present) without petrographic analysis.

We believe NRC should be using the same exact testing that was used in Belgium to identify embrittlement microcracking in all reactors, including Limerick, before they leak radioactive water.

Independent 3rd party testing experts need to do nuclear reactor embrittlement testing. Exelon

should NOT design the testing protocol nor hire the testing company for Limerick, when Exelon has such a huge vested interest in the outcome of this testing.

We Object To Strategies Used By NRC/Exelon In Lieu Of Testing

These alternatives allowed by NRC instead of actual embrittlement testing are not acceptable for Limerick.

- *Extrapolations – "guesstimates" using data that does not come from equipment in question*

  An example of a typical extrapolating: the decision to start up a Unit 1 reactor based on a Unit 2 inspection.
- *Acceptance of unsubstantiated self-serving reports by the licensee*
- *Announced NRC inspections can miss big problems*

  NRC inspectors identifying serious problems, fail to question the industry while conducting inspections within very narrow parameters defined by Exelon.
- *Aging effects prediction models based on a constant 5% rate into relicensing*

  Material fatigue increases with time, yet NRC is failing to acknowledge accelerated percentage rates for material fatigue.
- *Weld overlays to make cracked pipes thicker - a Band-Aide fix that is becoming routine.*

If Limerick Is Representative Of NRC's Regulatory Process, It Is Clear Our Nation's Nuclear Plants Are Too Risky. NRC Irresponsibly Exempted Exelon From Metal Fatigue Testing At Limerick, Even Though Safety Inspection Reports Revealed:

- *Wear and tear at BWR vessel attachments.*
- *Embrittlement of BWR components due to thermal aging.*
- *Boiling Water Reactor (BWR) stress corrosion cracking.*
- *Loss of material in steel caused by general pitting and crevice corrosion.*
- *Wall-thinning of carbon steel piping components exposed to reactor coolant (water).*
- *Loss of fracture toughness in some equipment.*
- *Loss of material in submerged bolting (loss of integrity).*
- *Loss of integrity in gray cast iron and copper.*
- *Loss of material and heat transfer in piping, their components and elements, heat exchangers, and tanks.*
- *Chemical contamination, corrosion, oxidation causing increased connection resistance in fuse holder metal clamps.*
- *Fatigue in fuse holders due to heating, thermal cycling, electrical transients, increased resistance of connection due to fatigue caused by frequent manipulation or vibration.*
- *Failure of Motor Operated Valve (MOV) system that delivers water to the core.*
- *Cracking or changes in piping exposed to raw water.*
- *Suppression pool in use "beyond its service life."*
- *Loss of material and cracking in spent fuel pool liner.*

It Is Alarming That NRC Has A History Of Weakening And Eliminating Material Fatigue Regulations For Limerick:

- *Exemptions Granted to Exelon for Limerick Include:*

  Exempted - Fatigue analysis based on Limerick's design code.

  Exempted - Fatigue analysis of control rod guide tube.
- *Relief Requests Granted to Exelon for Limerick, Including:*

  Relief Granted - Vessel Attachment Weld Inspection and Evaluation Guidelines

counting relief as compliance for re-licensing.

- *Changes Granted to Exelon for Limerick, Including:*
  1) *Core Operating Limits for Unit I granted April 3, 2013.*
  2) *Core Operating Limits for Unit 2 granted June 10, 2013.*
  3) *Reactor Pressure Boundary change granted.*
  4) *License Amendment to Modify Safety Limit Minimum Unit I granted, Jan. 30, 2012.*
- *Exclusions Provided to Exelon for Limerick:*

  Exclusion: NRC failed to require embrittlement testing for Limerick relicensing.
  Exclusion: NRC refused to address the substandard cement of Limerick's fuel pools as a condition for relicensing.
  Exclusion: Computer modeling and mere extrapolations are irresponsibly allowed, even though they have been proven woefully inadequate at predicting the actual severity of embrittlement in Japan.

Instead of requiring compliance with safety regulations and standards, NRC has relinquished safety control to the lobbying group for the nuclear industry (NEI), who in essence determines what regulations nuclear plants will comply with and which ones they will eliminate.

NRC has been rewriting its PTS standards, at least since 1982. When standards are violated, rather than requiring licensees to address embrittlement or shut down reactors, NRC simply weakens PTS standards, irresponsibly allowing severely embrittled nuclear reactors to keep operating, despite the risks for disaster.

It is obvious that embrittlement and cracking risks which we identified for Limerick's reactors require comprehensive independent testing. The issue is long-term nuclear plant stability and the avoidance of catastrophic meltdowns.

We believe that DG-1299 paves the way to the disaster that Belgium's regulators were trying to prevent when calling for embrittlement testing of nuclear plants world-wide.

We Urge NRC To Reconsider Its Current Efforts To Weaken Its Regulations Yet Again, Which Benefits The Nuclear Industry At Public Expense.

NRC Response to ACE-1

The NRC disagrees with the comments made in ACE-1 concerning the NRC's regulation of the Limerick Generating Station. In ACE-1 the commenter identified several concerns with the Limerick Generating Station in Pennsylvania, including reactor embrittlement, as examples of inadequate NRC regulation of the nuclear power industry. Similar comments have been raised by this commenter and addressed by the NRC in an on-going dialogue concerning the Limerick Power Station (ADAMS Accession No. ML15105A096).

It should be noted that most of the ACE-1 comments pertain to the Limerick Generating Station, which is a *boiling* water reactor (BWR), while DG-1299 concerns pressurized thermal shock, which is only a concern for pressurized water reactors (PWR). However, ACE-1 does raise several general points pertaining to DG-1299 to which the NRC offers the following responses:

- **Comment ACE-1-1**: *DG-1299 and accompanying draft NUREG-2163 … would weaken*

nuclear reactor embrittlement standards.

- **NRC Response**: The NRC disagrees with the comment. The NRC notes that the comment does not include examples or explanations of how DG-1299 or NUREG-2163 would weaken nuclear reactor embrittlement standards. Furthermore, DG-1299 is a draft regulatory guide. Regulatory guides establish methods that are acceptable to the NRC staff to demonstrate compliance with regulatory standards. However, regulatory guides do not establish any standards or requirements. As such, the provisions of DG-1299, if finalized, would not change reactor embrittlement standards or requirements established in applicable regulations.

- **Comment ACE-1-2**: *NRC's draft Regulatory Guide proposal, in lieu of embrittlement testing to prove safety, is indefensible. New understandings about embrittlement suggest that specific embrittlement testing is the only way to accurately determine the safety condition of Limerick's reactor structures and equipment.*
  - **NRC Response**: The NRC disagrees with the comment. 10 CFR 50.61a(f)(6) requires consideration and evaluation of plant-specific embrittlement test data. Thus, DG-1299, which outlines methods by which this consideration and evaluation may be performed, supports embrittlement testing and is not a proposal that has been made *"in lieu of embrittlement testing."*
- **Comment ACE-1-3**: *Testing recommended by some nuclear experts to accurately identify embrittlement and microcracking includes ultrasonic testing …* [and testing of] *metal coupons (in the RPV)."*
  - **NRC Response**: The NRC disagrees with the comment. 10 CFR 50.61a(e) requires ultrasonic testing. DG-1299 outlines methods acceptable to the staff by which this testing and evaluation may be performed. 10 CFR 50.61a does not require testing of metal coupons (surveillance samples); these requirements are established as part of a separate regulation: 10 CFR Part 50, Appendix H. However, 10 CFR 50.61a(f)(6)(i) requires evaluation of surveillance data that is available consistent with the requirements established in 10 CFR Part 50, Appendix H. DG-1299 outlines methods acceptable to the staff by which statistical analysis of the surveillance coupon data may be performed.
- **Comment ACE-1-4**: *NRC has been revising its PTS standards since 1982. When standards are violated, rather than requiring licensees to address embrittlement or shut down reactors, NRC simply weakens PTS standards, irresponsibly allowing severely embrittled nuclear reactors to keep operating, despite the risks for disaster.*
  - **NRC Response**: The NRC disagrees with the comment. Regulations concerning PTS have been written twice: In 1985 (50 FR 29944) and 2010 (75 FR 23). As discussed in NUREG-1806 and NUREG-1874 (ADAMS Accession Nos. ML072830074 and ML061580343, respectively), the risk level associated with the embrittlement limits of the 2010 PTS standard is actually five times more stringent than that of the 1985 standard. Thus, the comment that PTS regulations have been weakened over time is incorrect. Also, the comment that PTS standards have been violated is incorrect. All plants have remained compliant with the provisions of the PTS rule (10 CFR 50.61) or they have taken appropriate remediative actions as outlined in that rule. Finally, it should be noted that, as a regulatory guide, DG-1299 does not establish any standards concerning PTS.
- **Comment ACE-1-5**: *We believe that DG-1299 paves the way to the disaster that Belgium's regulators were trying to prevent when calling for embrittlement testing of nuclear plants world-wide.*
  - **NRC Response**: The NRC disagrees with the comment. The NRC is aware of the

recent operational experience concerning the Doel 3 and Tihange 2 reactors in Belgium (see NRC Information Notice 2013-19, "Quasi-Laminar Indications in Reactor Pressure Vessel Forgings, September 22, 2013, ADAMS Accession No. ML13242A263). The staff continues to review all information that is made available by the Belgian regulatory agency (FANC) on this topic, and assess the potential impact of this information on plants operating in the United States. Based on currently available information, the NRC concludes that no action is needed in response to this operational experience from Belgium because industry investigations for the U.S. plants, which are being evaluated and validated by the NRC and its contractors, do not indicate the presence of similar conditions in the fleet of U.S. reactors. This evaluation will be updated in the future as new information becomes available, and would be considered in a future revision of DG-1299, if necessary.

No changes were made to the DG or NUREG in response to comments ACE-1-1 through ACE-1-5.

Comment PWROG-2-1

Guidance on Equation (5) limitations/use parameters are not stated in DG-1299. Add wording similar to NUREG-2163 on using caution when applying the equation to conditions near to or beyond the 5 extremes of its calibration dataset. Could also include information in Table 5 of NUREG-2163 in the Reg. Guide itself.

NRC Response to PWROG-2-1

The NRC agrees with the comment. Adding information to the DG concerning limitations on the use of Equation (5) will provide clarity concerning the range of variables to which the embrittlement trend curve equation was calibrated. To that end, the following text and table will be added immediately below existing Table 1 in the DG:

As with any equation calibrated to empirical data, inaccuracies have a greater tendency to occur at the extremes, or beyond the limits, of the calibration dataset. Users of Eqn. (5) should therefore exercise caution when applying it to conditions near to or beyond the extremes of its calibration dataset, which appear in Table 2.

Table 2. Independent Variables in the Eqn. (5) ETC and the Ranges and Mean Values of the Calibration Dataset

Variable	Symbol	Units	Values of Surveillance Database			
			Average	Standard Deviation	Minimum	Maximum
Neutron Fluence (E > 1 MeV)	ϕt	n/cm^2	1.24E+19	1.19E+19	9.26E+15	1.07E+20
Neutron Flux (E > 1 MeV)	ϕ	$n/cm^2/sec$	8.69E+10	9.96E+10	2.62E+08	1.63E+12
Irradiation Temperature	T_C	°F	545	11	522	570
Copper Content	Cu	weight %	0.140	0.084	0.010	0.410
Nickel Content	Ni	weight %	0.56	0.23	0.04	1.26
Manganese Content	Mn	weight %	1.31	0.26	0.58	1.96

Phosphorus Content	*P*	*weight %*	*0.012*	*0.004*	*0.003*	*0.031*

Existing Tables 2 through 4 and their citations in the text of the DG will be re-numbered accordingly.

The changes noted above were made to the DG in response to this comment. No changes were made to the NUREG in response to this comment.

Comment PWROG-2-2

Page 3, #1. Editorial error on line 9 of item #1. "10 CFR 50.61a criteria Such applicants."

NRC Response to PWROG-2-2

The NRC agrees with the comment. A period was inserted after the word "criteria" in the DG in response to this comment. No changes were made to the NUREG in response to this comment.

Comment PWROG-2-3

Page 6, Step 1, (a). "For each shell material in the RPV beltline region, identify all surveillance data from the plant being assessed and from any other reactor that is operating, or has previously operated, under a license issued by the NRC that is of the same heat of material." Inclusion of BWR capsule data for the purposes of assessing materials in PWR plants typically does not provide relevant results due to the lower accrued fluence/flux in BWR environments. Recommend replacing "reactor" with "PWR."

NRC Response to PWROG-2-3

The NRC disagrees with the comment. The staff considered this matter when developing this regulatory guidance. The comment is correct that BWR surveillance data typically has a much lower fluence than PWR surveillance data. However, because the quantity of surveillance data for individual material heats is limited, valuable information pertinent to the material being evaluated could be overlooked by an *a priori* exclusion of BWR data. Furthermore, the embrittlement trend curve (ETC) developed in 10 CFR 50.61a was based on evaluation of all surveillance data, BWR and PWR alike. Therefore, it is appropriate and consistent with the ETC development to consider BWR data. The staff also recognizes that individual licensees may elect to exclude other plant data in their plant-specific assessments, for reasons similar to those given in the comment, and they may do so with adequate technical justification. Considering this, the staff carefully chose the words, "…should ***assess*** its surveillance data…" rather than "…should ***include*** its surveillance data…" [***emphasis added***] at the beginning of Step 1 on page 6 in the DG.

Additionally, the NRC notes that guidance concerning an outlier test failure due to spurious data at low fluence applies equally to BWR surveillance data.

For these reasons, no changes were made to the DG or the NUREG in response to this comment.

Comment PWROG-2-4

Page 7, Step 1, (c). NRC lists "Charpy-V notch energy data used to estimate DT30" as an item to be assembled. The energy data is not essential information for the purposes of performing the statistical analysis, just the resulting ΔT_{30}.

NRC Response to PWROG-2-4

The NRC disagrees with the comment. While the ΔT_{30} data are all that are needed to perform the statistical analyses required by 10 CFR 50.61a, the ΔT_{30} data are derived from energy data. The Charpy energy data are useful both for the purposes of ensuring ΔT_{30} is calculated correctly and for situations where unusual or unexpected trends need to be diagnosed.

No changes were made to the DG or NUREG in response to this comment.

Comment PWROG-2-5

Page 7, Step 1, (c). T_c is defined as the time-weighted average "from the start of full power operation through the end of licensed operation." This definition is consistent with the one provided in 10 CFR 50.61a, however, for the purposes of the statistical check, this definition is not entirely appropriate. It would not be appropriate to apply a time weighted temperature for the entire operation of the plant on a surveillance capsule that was not subjected to that environment. T_c should be defined as the time weighted average coolant temperature of the reactor coolant system cool leg covering the inservice operation time that a surveillance capsule is subjected to.

NRC Response to PWROG-2-5

The NRC agrees with the comment. The value of Tc used in a statistical check should reflect the temperature the Charpy specimens in the surveillance capsule have seen, as reflected by this comment. The definition of T_c on page 7 of the DG and page 25 of the NUREG will therefore be modified as follows (changes appear in ***underlined boldface font*** *below*):

Note: T_C (°F) is determined as the time-weighted average coolant temperature of the reactor coolant system ***cold*** *leg covering the time period* ***when the capsule was in the reactor*** ~~***from the start of full power operation through the end of licensed operation***~~.

Comment PWROG-2-6

Page 7, T_c bullet. Difference between time-weighted temperature vs. pure average (mean) temperature has been found to be quite minimal for plants analyzed to date. Practical observation.

NRC Response to PWROG-2-6

The NRC agrees with the comment. No changes were made to the DG or NUREG in

response to this comment.

Comment EPRI-3-1

DG-1299, Page 3, Item 1, "Construction Requirements." This item indicates that licensees must apply for an exemption to Item (b) of 10 CFR 50.61a (hereafter, "the rule") if it is used for vessels fabricated subsequent to a certain date. A regulatory guide, which is not a substitute for a rule and for which compliance is not required, should not be used to encourage or require exemptions to the rule. Item (b) of the rule should be modified so that a licensee wishing to use the rule for a vessel fabricated subsequent to a certain date submits a request and justification for implementing the rule to the Director of NRR for review and approval. This change will allow approval in accordance with the rule rather than an exemption to the rule.

NRC Response to EPRI 3-1

The NRC agrees, in part, with the comment. The NRC agrees that neither this, nor any, DG or RG should be used to encourage or require exemptions from a regulation. Nevertheless, there are certain situations where an exemption is required to use 10 CFR 50.61a, and that is the case, as specified in 10 CFR 50.61a(b), for nuclear power reactors whose construction permit was issued after February 3, 2010. As the Commission explained in the § 50.61a final rule statement of considerations, future applicants "may … seek an exemption from § 50.61a(b) to apply this rule…." "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," 75 Fed. Reg. 13, 14 (January 4, 2010). The NRC does not agree with the comment that Item (b) of the rule should be modified so that a licensee wishing to use the rule for a vessel fabricated subsequent to a certain date submits a request and justification for implementing the rule to the Director of NRR for review and approval. The technical basis for 10 CFR 50.61a does not support the application of the rule to newer vessels without additional technical support. Such support could be provided by a licensee as part of an exemption request

Concerning the former comment, to better clarify that the DG is ***not*** being used to encourage or require exemptions to the rule the DG and supporting tech basis NUREG will be changed as follows (changes appear in ***underlined boldface font*** below):

Changes to wording on page 3 of the DG:

> *Licensees whose construction permits were issued after February 3, 2010, or with reactor vessels that were not designed and fabricated to the 1998 Edition or earlier of the ASME code must apply for and receive a specific exemption from 10 CFR 50.61a(b) via 10 CFR 50.12 in order to utilize the alternate 10 CFR 50.61a criteria* ~~***Such applicants for an exemption should demonstrate that the risk-significant factors controlling PTS are adequately addressed by the technical basis calculations developed in support of the Alternate PTS Rule. Position 1 of this document identifies factors to be considered in such an evaluation.***~~.

Changes to wording on pages 4 and 5 of the DG:

> *If a licensee does not fit within this category (e.g., a licensee whose construction permit*

was issued after February 3, 2010), the provisions of 10 CFR 50.61a may not be used as an alternative to the requirements of 10 CFR 50.61 unless the licensee applies for and obtains a specific exemption under 10 CFR 50.12 from the 10 CFR 50.61a(b) prohibition. ***~~The criteria for obtaining such an exemption are listed in 10 CFR 50.12(a): the exemption must be authorized by law, must not present an undue risk to the public health and safety, must be consistent with the common defense and security, and special circumstances must be present. When addressing these exemption criteria, the licensee should demonstrate that the risk-significant factors controlling PTS for the plant in question are adequately addressed by the technical basis calculations that were performed to develop 10 CFR 50.61a. Factors to be considered in this evaluation should include the following:~~***

- ***~~The event sequences, which may lead to over-cooling of the RPV.~~***
- ***~~The thermal-hydraulic response of the nuclear steam supply system (NSSS) in response to such sequences.~~***
- ***~~Characteristics of the RPV design (e.g., vessel diameter, vessel wall thickness, operating pressure) that influence the stresses that develop in the beltline region of the vessel in response to the event sequences.~~***
    - ***~~Note: As indicated in Section 1.2 of NUREG-2163, the "reactor vessel beltline" is defined as those reactor vessel shell materials with projected neutron fluence values equal to or greater than 1 x 1017 n/cm2 at the end of the design life. Fluence values should be determined in accordance with methodology consistent with that specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, or using methods otherwise acceptable to the staff.~~***
- ***~~Characteristics of the RPV material and its embrittlement behavior.~~***

~~The technical details of how these factors were considered in the development of the Alternate PTS Rule are contained in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (Ref. 6).~~

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Changes to wording on page 7 of the NUREG:

…These screening criteria effectively define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific compensatory action or analysis ***~~unless the licensee receives an exemption from the requirements of 10 CFR 50.61.~~***

Comment EPRI-3-2

DG-1299, Page 3 through page 20. Most of the text in the draft guide (DG) is a repeat of the requirements in the rule. Text that repeats the rule is not clearly differentiated from the text that offers guidance for those conditions where the rule allows alternate procedures to demonstrate compliance with the rule. Adopt a consistent, uniform convention throughout the DG for differentiating (identifying) what is from the Rule and what is guidance. The text in Paragraph 4,

"Alternate Limits on Embrittlement" from the middle of page 18 to the top of page 19 illustrates an effective, succinct and easy to understand format that should be used throughout the RG. In this instance, the text identifies: the specific paragraphs in the rule where explicit compliance cannot be demonstrated, the specific paragraphs in the rule that allows alternate measures to be used to demonstrate compliance with the rule, and guidelines that are acceptable to the staff for demonstrating compliance to the rule. It is recommended that this same format be used to address the "Plant Specific Surveillance" and "ISI Data and NDE Requirements" sections.

NRC Response to EPRI-3-2

The NRC agrees, in part, with the comment. The NRC agrees that the regulation is referenced throughout the DG. The location of regulatory references within the various regulatory positions of the DG relative to the subsequent steps is intended to establish a clear link between the regulatory requirement in the rule and the related regulatory position in the DG. That is, by having the regulatory requirement preceding the regulatory position, a clear relationship is drawn between the requirement and the subsequently numbered regulatory position. This approach is consistent with NRC guidelines for preparation of regulatory guides. NRC has verified, and corrected where appropriate, the position of the regulatory reference as related to the subsequent supporting regulatory position in the DG.

No changes were made to the DG or NUREG in response to this comment.

Comment EPRI-3-3

DG-1299, Page 17, Figure 2. The flow chart in Figure 2 is difficult to understand and can be simplified. The flow chart in Figure 2 is unnecessary [sic] complex and can be made easier to understand by revising it to follow the flow described in Paragraph (e) (4) of the rule.

NRC Response to EPRI-3-3

The NRC disagrees with the comment. Paragraph (e)(4) of 10 CFR 50.61a encompasses only some of the requirements addressed by the guidance in the DG. The flow chart in Figure 2 addresses all of the requirements in 10 CFR 50.61a for which guidance is provided in the DG. Therefore, it would not be complete to limit Figure 2 to just those requirements in paragraph (e)(4).

However, in response to this comment, reference to the appropriate paragraph numbers from 10 CFR 50.61a will be added to each of the orange-colored "Step" blocks in the flow chart to clarify where the user can find the specific requirements being addressed for each step of the guidance. These same changes will also be made to Figure 5 of the NUREG. In addition, the following change will be made to page 41 of the NUREG at line 36 (changes appear in ***underlined boldface font*** below):

> *A summary of the process depicted in Figure 5 is as follows~~:~~. **For each step in Figure 5, reference is made to the applicable paragraph of the Rule that contains the requirement that is addressed by that step.***

Comment EPRI-3-4

DG-1299, Step I: text on page 15 and Figure 2 on Page 17. The block labeled "Evaluate for Acceptability" in Figure 2 indicates that either of two criteria can be used. However, the "Preclude Brittle Fracture" option is not mentioned in the rule and this option is not mentioned in the text in Step I. The text in Step I should be consistent with the flow chart in Figure 2 and identify the "Preclude Brittle Fracture" option and should state explicitly that compliance with the "Preclude Brittle Fracture" option provides assurance that TWCF < 1E-6.

NRC Response to EPRI-3-4

The NRC agrees, in part, with the comment. The NRC disagrees with the first portion of the comment, which indicates that the "Preclude Brittle Fracture" option is not mentioned in the text in Step I (note that Step I is now referred to as Step G due to editing. Under Step G on page 16 of the DG, Item 1, "Preclusion of brittle fracture," is included. The second option for Step G, which is indicated as "Calculate TWCF" in Figure 2 on page 17 of the DG, is reflected under Step G, Item 2, "Calculate the plant-specific TWCF using a plant-specific PFM analysis" on page 16 of the DG. Therefore, the text and Figure 2 of the DG are consistent with each other.

The NRC agrees with the second portion of the comment that the DG should state explicitly that compliance with the "Preclude Brittle Fracture" option provides assurance that TWCF < 1E-6. In response to this portion of the comment, the following will be added to Step I on page 15 of the DG (changes appear in ***underlined boldface font*** below):

> *Step G**: ~~If the results of Step H are not acceptable,~~ A demonstration that the TWCF is less than 1x10^{-6} events per reactor year is necessary to satisfy paragraph (e)(4) of the Alternate PTS Rule. The staff considers the two approaches detailed here to be acceptable for providing assurance that the TWCF is less than 1x10^{-6} events per reactor year. Therefore**, all flaws should be evaluated for acceptability using one of the following approaches:*

The same change will be made to the text at the top of page 43 of the NUREG.

Comment EPRI-3-5

DG-1299, Step G: text on page 15 and Figure 2 on Page 17. The block labeled "Account for NDE Uncertainty" in Figure 2 indicates that any or all of three procedures may be used to account for uncertainty. However, the text in Step G implies that all three items should be included to account for uncertainty. The text in Step G should be modified to be consistent with the "Account for NDE Uncertainty" block in Figure 2 and with NUREG-2163. It is recommended that the text in Step G be modified to include the following sentence from Section 6.4.2 of NUREG-2163: Any or all of these uncertainties may be considered, depending on the level of detail needed for flaw assessment.

NRC Response to EPRI-3-5

The NRC agrees with the comment. The text for Step G on page 15 of the DG will be changed as follows (changes appear in ***underlined boldface font*** below):

> *Step G: If the results of Step D are not acceptable, NDE uncertainties **such as flaw***

sizing errors, a flaw detection threshold, or probability of detection (POD) *may be accounted for in the evaluation. Appendix C of NUREG-2163 describes the development and application of one methodology acceptable to the NRC that accounts for uncertainties in NDE data. This method may be used for the purpose of developing more realistic vessel-specific flaw depth and density distributions for comparison to Tables 2 and 3 of 10 CFR 50.61a, as well as for use in a plant-specific PFM analysis. The methodology considers flaw sizing errors, a flaw detection threshold, ~~**probability of detection (**~~(POD~~**)**~~, and a prior flaw distribution assumption. It uses a Bayesian updating methodology to combine the observed NDE data with the available flaw data and models used as part of the PTS re-evaluation effort. The licensee must submit the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in (c)(2) of 10 CFR 50.61a.*

The same change will be made to the text for Step G on page 42 of the NUREG.

Comment EPRI-3-6

DG-1299, Page 18, First full paragraph, 4th line. The statement is made that this alternate procedure fulfills the requirements of (d)(3) through (d)(6) of the rule. It would appear from the structure of the rule that this alternative procedure would not fulfill (d)(3). The text in the RG should indicate that the alternate procedure fulfills requirements (d)(4) through (d)(6).

NRC Response to EPRI-3-6

The NRC disagrees with the comment. Paragraphs (d)(4), (d)(5), and (d)(6) of 10 CFR 50.61a all refer to the requirements of paragraph (d)(3), so the inclusion of the requirements in paragraph (d)(3) are relevant to the statement on page 18 of the DG.

No changes were made to the DG or NUREG in response to this comment.

Comment EPRI-3-7

DG-1299, Page 16, Paragraphs 2i and 2ii. These paragraphs refer to two computer codes developed with NRC funding to provide a posterior flaw distribution based on a prior flaw distribution and flaws detected by NDE examinations. These computer codes are not generally available. These computer codes should be made publically available for industry use in the same manner the FAVOR software is publically available.

NRC Response to EPRI-3-7

The NRC agrees with the comment. There are three computer codes referred to on page 16 of the DG. The first computer code is the software for performing a Bayesian update of the flaw distribution. A listing of a sample computer code is included in NUREG-2163 starting on page C-39 ("Appendix C-1: Sample MATLAB Routine"). The second computer code is FAVOR ("Fracture Analysis of Vessels - Oak Ridge"), and is available upon request from the authors of NUREG-2163. The third computer code is the generalized procedure for generating flaw-related inputs for the FAVOR Code described in NUREG/CR-6817 ("VFLAW"), as described on page 49 of NUREG-2163. VFLAW is available upon request from the authors of NUREG-2163.

No changes were made to the DG or NUREG in response to this comment.

Comment EPRI-3-8

DG-1299, Page 6, Section 2, Step 1, (a). In the application of RG 1.99, Revision 2, and 10 CFR 50.61, assessment and consideration of surveillance data has been limited to data from equivalent reactor types. In other words, BWR surveillance data has not been used for PWRs and vice versa. It is understood that the ETC contained in 10 CFR 50.61a has flux and temperature terms that allow for the consideration of reactor type. However, in some cases, low fluence BWR data is not appropriate for comparison with higher fluence PWR data. Consistent with the guidance in DG-1299, C.2, Step 2, (d), iii, 3, a, BWR data should be excluded from consideration in the surveillance data checks if it is at a fluence that is less than 10% of the fluence at which the PTS evaluation is being performed. The requirements for consideration of low fluence data from BWRs should be clarified and exclusion of such data should be permitted if it is for fluence levels less than 10% of the fluence at which the PTS evaluation is being performed.

NRC Response to EPRI-3-8

See NRC response to Comment PWROG-2-3.

Comment EPRI-3-9

DG-1299, Page 7, Section 2, Step 1, (c). The note for the coolant temperature explains that T_c is the time weighted average from the start of full power operation through the end of licensed operation. For the purposes of the surveillance data statistical analysis, this should be the average up until date that the capsule was withdrawn. Change "through the end of licensed operation" to "to the date of surveillance capsule withdrawal."

NRC Response to EPRI-3-9

See NRC response to Comment PWROG-2-5.

Comment EPRI-3-10

DG-1299, Page 7 Section 2, Step 1, (c). The text at the end of this section says that conservative estimates (mean plus one standard deviation) can be used. Suggest that it be identified that the conservative wt% values for P and Mn in Table 4 of 10 CFR 50.61a are acceptable for use. Add a sentence to the end of the paragraph stating that Table 4 of 10 CFR 50.61a provides conservative wt/% values for P and Mn that are acceptable for use in the statistical assessments in the absence of plant specific values.

NRC Response to EPRI-3-10

The NRC agrees with the comment. Making this clarification ensures that the guidance of the DG is consistent with the language of the rule. The following two sentences will therefore be added at the end of Step 1(c) on page 7 of the DG:

"Such conservative estimates for phosphorus and manganese appear in Table 4 of 10 CFR 50.61a. Similarly, upper bound estimates for Cu and Ni are provided in 10 CFR 50.61."

No changes were made to the NUREG in response to this comment.

Comment EPRI-3-11

DG-1299, Page 8, Section 2, Step 2, (a), iv. The text states that if the mean test is not satisfied you should "proceed to Step 2(d)." This is misleading because even if the mean test is failed, the slope and outlier test still need to be considered. Delete "; in this case proceed to Step 2(d)."

NRC Response to EPRI-3-11

The NRC agrees with the comment. As explained in the DG, section C.2, Step 2(d)(ii), licensees need to perform all three statistical tests even when only one of the mean, slope, and outlier tests is not satisfied.

The text on page 8 of the DG will be revised as follows (changes appear in ***underlined boldface font*** below):

iv. If r_{mean} exceeds r_{max}, then the mean test is not satisfied~~; in this case proceed to Step 2(d)~~. If r_{mean} is less than or equal to r_{max}, then the mean test is satisfied; in this case proceed to Step 2(b).

No changes were made to the NUREG in response to this comment.

Comment EPRI-3-12

DG-1299, Page 8, Section 2, Table 2, footnote a. It is not clear what the purpose of footnote "a" is in the RG. Delete footnote "a" from Table 2, or clarify its purpose.

NRC Response to EPRI-3-12

The NRC agrees with the comment. Footnote "a" indicated that the standard deviation of *21.2, which represents the uncertainty in the ΔT_{30} values,*] included the standard reference materials. The rule does not require the analysis of so-called "standard reference materials" so the value associated with their standard deviation is not needed.

Footnote "a" of Table 2 on page 8 of the DG, and also of Table 4 on page 23 of the NUREG, will be deleted.

Comment EPRI-3-13

DG-1299, Page 9, Section 2, Step 2, (b), iv. The text states that if the slope test is not satisfied you should "proceed to Step 2(d)." This is misleading because even if the mean test is failed,

the outlier test still needs to be considered. Delete "; in this case proceed to Step 2(d)."

NRC Response to EPRI-3-13

The NRC agrees with the comment. As explained in DG-1299, section C.2, Step 2(d)(ii), licensees need to consider test data from all three tests even when one of the mean, slope, and outlier tests is not satisfied.

The text on page 9 of the DG will be revised as follows (changes appear in ***underlined boldface font*** below):

> *iv. If T_{SURV} exceeds T_{CRIT}, then the slope test is not satisfied~~; in this case proceed to Step 2(d)~~. If T_{SURV} is less than or equal to T_{CRIT}**,** then the slope test is satisfied; in this case proceed to Step 2(c).*

No changes were made to the NUREG in response to this comment.

Comment EPRI-3-14

DG-1299, Page 17, Section 3, Figure 2 and NUREG-2163, Page 45, Section 6.1, Figure 5. In step D, the meaning or intent of the statement, "Remaining Flaws Must be Acceptable per ASME, Section XI, Table IWB-3510-1" is not clear. Furthermore, this is not discussed in the text for Step D on page 15. Add clarification that the "Remaining flaws" are those flaws in the inner 3/8 vessel thickness that were not evaluated in Step F.

NRC Response to EPRI-3-14

The NRC agrees with the comment. The current text is, indeed, not clear. To clarity, the text, "Remaining Flaws must be Acceptable per ASME, Section XI, Table IWB-3510-1" that appears in the box adjacent to "Step D" in Figure 2 of the DG and Figure 5 of the NUREG will be deleted. The text is not necessary because it is a prerequisite from Step C that all flaws evaluated in Step D must be acceptable per ASME Code, Section XI, Table IWB-3510-1.

Comment EPRI-3-15

DG-1299, Page 3, line 1. States "The 'Alternate PTS Rule' contained in 10 CFR 50.61a is revised PTS screening criteria...." Change to "The 'Alternate PTS Rule' contained in 10 CFR 50.61a provides revised PTS screening criteria...."

NRC Response to EPRI-3-15

The NRC agrees with the comment. The use of "is" rather than "provides," as suggested by the comment, is a typographical error.

The text on page 3 of the DG will be revised as follows (changes appear in ***underlined boldface font*** below):

> *The "Alternate PTS Rule" contained in 10 CFR 50.61a ~~is~~ **provides** revised PTS*

screening criteria in the form of an embrittlement reference temperature, RT_{MAX-X}, which characterizes the RPV material's resistance to fracture initiating from flaws based on more comprehensive analysis methods.

No changes were made to the NUREG in response to this comment.

Comment EPRI-3-16

DG-1299, Page 3, regulatory position #1, line 9. The period at the end of the sentence is missing. Add period after "criteria."

NRC Response to EPRI-3-16

See NRC response to Comment PWROG-2-2.

Comment EPRI-3-17

DG-1299, Page 18, Section 3, Figure 3 and NUREG-2163, Page 55, Section 6.3, Figure 6. This figure is confusing and it is not clear what value it adds beyond what is shown in Figure 2 (DG-1299) and Figure 5 (NUREG-2163). Step A, "Combine Recordable Flaws" is redundant with the box that says "Section XI Flaw Proximity Criteria" and it implies that flaws are required to be further combined for use with the 10 CFR 50.61a flaw limits, which is not the case. The labeling for Step F should be isolated to the "Section XI IWB-3600 Flaw Evaluation" decision point. The use of Tables 2 and 3 of 10 CFR 50.61a should be shown to the right of the figure and receive input from the "Acceptable, Use As-Is" and "Acceptable: Operate for Time Interval, Augmented Examinations" boxes. Delete the figure or revise as discussed in the comment.

NRC Response to EPRI-3-17

The NRC agrees with the comment. As originally printed the figure is not clear. To improve the clarity of Figure 4 (note, Figure 3 became Figure 4 due to editing) of the DG and Figure 6 of the NUREG they will both be revised as follows:

- The red solid-line box and associated label "10 CFR 50.61a Process" will be deleted.
- The red dotted-line box and associated label "Step F" will be deleted.
- The "Acceptable: Operate for Time Interval, Augmented Examinations" box and the "YES" arrow leading to it will be deleted.
- The "Unacceptable: Repair or Replace" box and the "NO (Defects)" arrow leading to it will be deleted.
- The "Section XI IWB-3600 Flaw Evaluation (Acceptable?)" diamond will be revised to a box with the title, "Evaluation in Accordance with ASME Code, Section XI is Required."
- The "Step A Combined Recordable Flaws" yellow oval will be deleted.
- "(Combine recordable flaws, if necessary)" will be added to the "Section XI Flaw Proximity Criteria" box.
- The "Plant-specific NDE data for comparison to Tables 2 and 3 in 10 CFR 50.61a" box will be moved to the right side of the figure and receive input from the "Section XI Flaw Proximity Criteria" and the "Acceptable: Use As-Is" boxes.

Comment EPRI-3-18

DG-1299, Page 18, Section 4, Step 3. Not all flaw populations may be appropriate for a given vessel. Insert "(as applicable)" following "and forging flaw populations."

NRC Response to EPRI-3-18

The NRC agrees with the comment. As an example, it would be inappropriate to assess vessels constructed from plates and welds using a flaw population appropriate for forgings. For this reason, the text on page 18 of the DG will be revised as follows (changes appear in ***underlined boldface font*** below):

- *Step 3. Estimate the 95th percentile TWCF value, $TWCF_{95\text{-}XX}$, for each of the axial weld flaw, plate flaw, circumferential weld flaw, and forging flaw populations **(as applicable)** using the $RT_{MAX\text{-}X}$ values from Step 2 and the formulae given in Section 3.5.1 of NUREG-1874.*

In addition, the following change will be made in Section 7 on page 67 of the NUREG (changes appear in ***underlined boldface font*** below):

1. *Determine $RT_{MAX\text{-}X}$ for all axial welds ($RT_{MAX\text{-}AW}$), plates ($RT_{MAX\text{-}PL}$), circumferential welds ($RT_{MAX\text{-}CW}$), and forgings ($RT_{MAX\text{-}FO}$) **(as applicable)** in the RPV beltline region according to the requirements of 10 CFR 50.61a. These $RT_{MAX\text{-}X}$ values must be expressed in units of Rankine (R) (degrees Fahrenheit (°F) plus 459.69).*

Comment EPRI-3-19

DG-1299, Page 6, Equation for $g(Cu_e,Ni,\varphi t_e)$. Incorrect symbol is used to denote multiplication. The symbol between "0.5" and "tanh" is the variable x, it should be "×," which is the symbol used to indicate multiplication elsewhere in the equations. Replace x with "×."

NRC Response to EPRI-3-19

The NRC agrees with the comment. The italicized *x* has been replaced with the multiplication symbol × in this equation in the DG, and also in the equation on page 22 of the NUREG.

Comment EPRI-3-20

DG-1299, Page 7, item (c). The list of information to be assembled is missing flux. Add flux to the list of information to be assembled.

NRC Response to EPRI-3-20

This NRC agrees with the comment. Flux needs to be included in this list because it is an important descriptor of the neutron environment to which the materials of the reactor are

exposed. “Flux” will therefore be added to the list of information in item (c) on page 7 of the DG. The same addition will be made to the list of information on page 25 of the NUREG.

Comment EPRI-3-21

DG-1299, Pages 9 and 10. The steps for the Outlier test are numbered incorrectly. On page 10, “Table 4” is erroneously numbered as step “iii.” Correctly number the steps associated with the Outlier test.

NRC Response to EPRI-3-21

This NRC agrees with the comment. The content of Step iii in the DG (“Table 4”) will be appended to the end of the last sentence of Step ii, and Step iv will be re-numbered to Step iii on pages 9 and 10 of the DG.

No changes were made to the NUREG in response to this comment.

Comment EPRI-3-22

NUREG-2163, Page xiii, Exec. Summary, Line 36. Insert the word “such” between “(ETC)” and “that.”

NRC Response to EPRI-3-22

The NRC agrees with the comment. The NUREG will be revised as suggested in the comment. No changes were made to the DG in response to this comment.

Comment EPRI-3-23

NUREG-2163, Page xvii. Definition for T_c should state that it is a time weighted average of the coolant temperature. Change definition to “Time weighted average of the Irradiated (coolant) temperature in degrees Fahrenheit (°F) or Celsius (°C).”

NRC Response to EPRI-3-23

See NRC response to Comment PWROG-2-5.

Comment EPRI-3-24

NUREG-2163, Page 3, Section 1.2, Lines 43-45. DG-1299 refers to this text and says that the beltline is defined as the “shell materials” with fluences projected to be greater than 1E17 n/cm^2. However, this text defines the beltline as “all regions of the RPV adjacent to the core” exposed to fluences greater than 1E17. The definition in the NUREG is confusing because “adjacent” is open to interpretation. The current text implies that the beltline is those materials that are both next to the core and have fluence projected to exceed 1E17. Make the definition of beltline in the NUREG consistent with that in DG-1299 by referring to the “shell materials” with projected neutron fluences equal to or greater than 1E17 n/cm^2.

NRC Response to EPRI-3-24

The NRC agrees with the comment. However, in response to Comment EPRI-3-1 of this document the subject text was deleted, rendering the comment moot.

Comment EPRI-3-25

NUREG-2163, Page 9, Section 2.4, Line 22. The limits of Table IWB-3510-1 only apply to flaws within the inner 3/8 thickness. Insert "within the inner 3/8 of the vessel thickness" between "flaws" and "exceeding."

NRC Response to EPRI-3-25

The NRC agrees with the comment. The NUREG will be revised as suggested in the comment. No changes were made to the DG in response to this comment.

Comment EPRI-3-26

NUREG-2163, Page 14, Section 3, Table 3, Item 8. The intent of this recommendation of the MRP was that all flaws detected in the ASME Section XI weld examination volume (weld width + 1/2T on either side of the weld) would be considered as weld flaws with the exception of those flaws on the very edges of the examination volume (those flaws 1/2T-0.5" from the weld). Those flaws would be considered plate flaws. The point of the recommendation was that HAZ flaws should be considered weld flaws. The NRC response indicates that this comment was agreed with. That is not entirely accurate. It is acknowledged in the footnote on page 53 that HAZ flaws may be considered as weld flaws. However, the MRP recommendation was not entirely implemented. NRC response for item 8 on page 14 should be revised to reflect partial agreement with the MRP recommendation.

NRC Response

The NRC agrees with the comment. The wording for the NRC Response to No. 8 in Table 3 on page 14 of the NUREG will be revised as follows (changes appear in ***underlined boldface font*** below):

> *The NRC agrees**, in part**, with this comment. Guidance for establishing whether the flaws are in the plate or the weld is included in this NUREG (see Step D in Section 6.~~1~~**3**).*

No changes were made to the DG in response to this comment.

Comment EPRI-3-27

NUREG-2163, Page 24, Section 5.2, 1, b). In the application of RG 1.99, Rev. 2, and 10 CFR 50.61, assessment and consideration of surveillance data has been limited to data from equivalent reactor types - e.g., BWR surveillance data has not been used for PWRs and vice versa. It is understood that the ETC contained in 10 CFR 50.61a has flux and temperature terms that may allow for consideration of shift data for materials irradiated in a different reactor type. However, in some cases, low fluence BWR data is not appropriate for comparison with higher fluence PWR data. Consistent with the guidance in DG-1299, 2, Step 2, (d), iii, 3, a,

BWR data should be excluded from consideration in the surveillance data checks if it is at a fluence that is less than 10% of the fluence at which the PTS evaluation is being performed. The requirements for consideration of low fluence data from BWRs should be clarified, and exclusion of such data should be permitted if fluence of the BWR data is less than 10% of the fluence at which the PTS evaluation is being performed.

NRC Response to EPRI-3-27

See NRC response to Comment PWROG-2-3.

Comment BN-4-1

My first comment is that I think there is an error on your <regulations.gov> online posting. The last sentence of II, Discussion reads:

"The alternate PTS requirements are based on updated analysis methods, and are desirable because the requirements in 10 CFR 50.61a are based on overly conservative probabilistic fracture mechanics analyses."

I think you meant to say 10 CFR 50.61, not 10 CFR 50.61a. After all, 10 CFR 50.61a is the alternate PTS rule itself, after all.

However, I must add that the phrase "overly conservative" strikes me as an odd one for NRC to use, and an odd justification or rationale for the regulatory relief, or safety regulation weakening, that 10 CFR 50.61a represents for an atomic reactor like Entergy Nuclear's Palisades in Michigan, which -- by both Entergy and NRC admission -- will violate 10 CFR 50.61 screening criteria by August, 2017. After all, NRC's supposed mission, mandate, and job is supposed to be to protect public health, safety, and the environment.

(Actually, by NRC and/or Palisades' -- be it Consumers Power, Consumers Energy, Nuclear Management Company, or Entergy Nuclear -- own admissions, Palisades' RPV "End of Life" due to RPV embrittlement/PTS risk had, previously, been set at 1995, then 1997, 1999, 2001, 2004, 2007, 2014, April 2017, and August 2017, all under 10CFR50.61. Now, under 10CFR50.61 a, NRC stands poised to permit Palisades to operate till 2031, despite its dangerously embrittled RPV. This "moving target" collusion to accommodate Palisades' continued operation is outrageous!).

But in this case, NRC seems to see its role as weakening regulations enough to accommodate a dangerously embrittled reactor pressure vessel, as at Palisades, so it can continue operating. NRC seems to see its role as allowing Palisades to come as close to an arbitrary "acceptably risky" line as possible. 50.61a allows for a 1 in a million risk, per reactor year, of a through-wall RPV fracture, due to pressurized thermal shock (PTS). NRC staff defends doing this, by stating that the NRC Commissioners have blessed this level of risk as "reasonable assurance of adequate protection." A growing number of people, such as those living in the shadow of Palisades, now understand that 50.61a means that NRC is fine with chipping away at "overly conservative" safety standards, at their expense. Chipping away at "overly conservative" safety standards, of course, means increasing risks.

The public downwind and downstream, up the food chain, and down the generations from

Palisades, and other badly embrittled RPVs -- such as Point Beach Unit 2, WI, also on the Lake Michigan shore, like Palisades, and, also like Palisades, predicted to surpass 50.61 safety standards by 2017 -- does not agree with NRC that a 1 in a million per reactor year risk of PTS through-wall cracking is "acceptably risky." It is not reasonable, adequate, protective enough, nor acceptable.

After all, what was the risk of the Fukushima Daiichi nuclear catastrophe happening, before it actually happened? Let's assume the risk, prior to 3/11/11, was 1 in a million per reactor year, for a meltdown and catastrophic radioactivity release at a reactor unit at Fukushima Daiichi. Therefore, the risk of three meltdowns and consequent radioactivity releases would have been 1 in a million X 1 in a million X 1 in a million = 1 in 1,000,000,000,000,000,000/reactor year. Truly, a very low probability risk. But it did happen, beginning on 3/11/11. And the consequences have been nightmarishly high, and will continue to be so for a very long time into the future (Cs-137 and Sr-90 in the living environment and food chain remain hazardous for 300-600 years; Pu-239 that escaped will remain hazardous for 240,000 to 480,000 years; 1-129 will remain hazardous for 157 million to 314 million years; etc.)

A 1 in a million per reactor year risk is much more likely to occur, than was the Fukushima Daiichi triple meltdown, and ongoing, catastrophic radioactivity release.

Another such example: what was the professional judgment as to PRA (probabilistic risk assessment) for the Space Shuttle Challenger to explode on lift off, before it actually did? 1 in 100,000? Well, that was off. The Space Shuttle Challenger was only the 25th such launch. So a 1/100,000 risk instantly became a 1/25 risk in one tragic moment.

We object to the over-reliance on PRA in 10CFR50.61a. We call for actual physical data to be returned to prominence in this regulatory realm. For example, at Palisades, four metal capsules remain available in the Palisades RPV for testing. And yet, the 2007 test was simply cancelled. A full 16 years will have passed between Palisades' last test, and its next scheduled (2003 to 2019). An over-reliance, an exclusive reliance, on PRA, conjecture, extrapolation, and projection, when substantial physical data remains available, is unacceptable. Pulling and testing capsules would deliver a dose of reality to these proceedings.

NRC Response to BN-4-1

The NRC agrees with the comment in part.

The NRC agrees that the typographical error described in paragraphs 1-3 of the comment exists. The Federal Register Notice dated March 13, 2015 (80 FR 13449) should have read, in part, as follows:

> *"The alternate PTS requirements are based on updated analysis methods, and are desirable because the requirements in **10 CFR 50.61** are based on overly conservative probabilistic fracture mechanics analyses."*

Concerning the comment in paragraph 4 on the use of the phrase "overly conservative," the NRC does not agree. The NRC notes that the development of 10 CFR 50.61a was undertaken in a manner consistent with the Commission's policy statement on the use of probabilistic risk assessment methods published on August 16, 1995 (60 FR 42622). This document states, in part, the following (***emphasis added***):

… the Commission adopts the following policy statement regarding the expanded NRC use of PRA:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

*(2) **PRA and associated analyses** (e.g., sensitivity studies, uncertainty analyses, and importance measures) **should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism** associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. …*

In the current context, the use of the phrase *"overly conservative"* in the March 13, 2015 *Federal Register* Notice (80 FR 13449) is consistent with the phrase *"reduce unnecessary conservatism"* used in the 1995 policy statement. As such, the development of 10 CFR 50.61a (and, therefore, DG-1299) is consistent with Commission policy and with the NRC's mission.

Concerning the comment in paragraph 6 that the NRC is *"chipping away at 'overly conservative' safety standards,"* it is important to understand the basis for the Commission's 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (60 FR 42622). In this document the Commission stated the following:

> *The policy statement addressed the need to remove unnecessary conservatism associated with regulatory requirements. It is not the Commission's intent to replace traditional defense-in-depth concepts with PRA, but rather to exploit the use of PRA insights to further understand the risk and improve risk-effective safety decision-making in regulatory matters. In doing so, the Commission is focusing its attention and resource allocation to areas of true safety significance.*

The remaining paragraphs of this comment do not directly pertain to any provision of DG-1299 or NUREG-2163. Rather, they pertain to either the requirements of 10 CFR 50.61a (which was noticed for public comment on October 3, 2007 [72 FR 56275] and August 11, 2008 [73 FR 46557]) or to broader policy issues. As such these comments lie beyond the scope of this request for public comment and are not addressed here.

No changes were made to either DG-1299 or to NUREG-2163 in response to this comment.

Comment BN-5-1

Please see the attached December 1, 2014 "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." (36 pages in total). Please accept the numerous challenges and criticisms contained therein, in the context of Entergy Nuclear's July 2014 License Amendment Request for 10CFR50.61a regulatory relief, as public comments in your DG-1299/NUREG-2163 proceeding. As indicated in the attached petition, this public comment is submitted on behalf of

Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future—Shoreline Chapter, and Nuclear Energy Information Service.

NRC Response to BN-5-1

The NRC disagrees with the comment. The referenced attachment concerns a license amendment request for the Palisades Nuclear Power Plant to use 10 CFR 50.61a (ADAMS Accession No. ML14335A807). This petition, which was amended on December 8, 2014 (ADAMS Accession No. ML14344A321), was the subject of hearings before the Atomic Safety Licensing Board. The matter was addressed by the Board in their Order issued on May 8, 2015 (ADAMS Accession No. ML15128A489) where it was concluded that, although the Petitioners demonstrated standing to intervene, they did not put forward an admissible contention. The Petitioners appealed their contention, and the appeal was rejected by the Commission.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-6-1

Please see the attached file, Declaration of Arnold Gunderson. Arnold Gundersen, Chief Engineer at Fairewinds Associates, Inc. in Burlington, VT, serves as the expert witness for Beyond Nuclear, Don't Waste MI, MSEF--Shoreline Chapter, and NEIS, in their coalition intervention against Entergy Nuclear's LAR at Palisades for 50.61a regulatory relief. Please accept the attached document, including its numerous challenges and critiques of 50.61a and Palisades' related LAR, as public comment on NRC's development of DG-1299 and NUREG-2163.

NRC Response to BN-6-1

See NRC response to Comment BN-5-1.

Comment BN-7-1

Please see the attached file, Declaration of Pierman, Kamps and Keegan Concerning Coupon Availability for PTS Testing, dated December 1, 2014. It was submitted as a part of the Beyond Nuclear-Don't Waste MI-MSEF Shoreline Chapter-NEIS intervention against Entergy Nuclear Palisades' LAR for 10CFR50.61a regulatory relief. Please accept this document as public comment in this proceeding.

We object to 10CFR50.61a's dismissing and ignoring existing, readily available hard physical data, and relying largely to exclusively instead on probabilistic risk assessment.

Palisades, for example, has four capsules remaining, which could be pulled and tested. But instead of accessing this readily available physical data, a scheduled 2007 capsule pull and test was simply canceled. A full 16 years between capsule pulls and tests (2003 to 2019) will have passed, with no reality check in between. Apparently, Entergy has no plans to access three capsules at all, during the 2019 to 2031 (extended license expiration in 2031) time period. This is unacceptable.

NRC's rationale, as documented in the attached affidavit, that physical capsules can't be pulled and tested, because then there would be no physical capsules left to pull and test, is absurd and unacceptable from a perspective of protecting public health, safety, and the environment from a scientific, physical reality-based regulatory approach. Over- to exclusive-reliance on PRA is unacceptable, and such aspects of 10CFR50.61a, as embodied in DG-1299 and NUREG-2163, cannot be allowed to stand in NRC regulations, regulatory guides, and their technical bases documents.

NRC Response

The NRC disagrees with the comment. The NRC response to Comment BN-5-1 of this document addresses the first paragraph of this comment.

Regarding the remaining paragraphs of this comment:

- DG-1299 reflects the requirements of 10 CFR 50.61a regarding consideration of physical test data from surveillance capsules. Specifically, 10 CFR 50.61a requires that data from capsules already tested be considered as part of a license amendment made in accordance with 10 CFR 50.61a. As such, the comment that the NRC is "*dismissing and ignoring existing, readily available hard physical data, and relying largely to exclusively instead on probabilistic risk assessment"* is incorrect.
- The schedule for surveillance capsule removal and testing is regulated by Appendix H to 10 CFR Part 50. This Appendix is not the subject of this request for public comments.

As such these paragraphs of the comment are beyond the scope of this request for public comment and are not addressed here.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-8-1

Please see the attached file, "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 10CFR50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (20 pages).

Please accept the numerous challenges and criticisms contained therein, in the context of Entergy Nuclear's July 2014 License Amendment Request for 10CFR50.61a regulatory relief, as public comments in your DG-1299/NUREG-2163 proceeding. As indicated in the attached COMBINED REPLY, these public comments are submitted on behalf of Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future—Shoreline Chapter, and Nuclear Energy Information Service.

NRC Response to BN-8-1

See NRC response to Comment BN-5-1.

Comment BN-9-1

Please see the attached files:

1. *Nuclear Reactor Pressure Vessel Crisis: Greenpeace Briefings (dated Feb. 15, 2015; 10 pages);*
2. *Greenpeace press release, "Thousands more cracks found in Belgian nuclear reactors, Belgian regulatory head warns of global implications," dateline Brussels, Feb. 17, 2015 (2 pages).*

As revealed in this report from Greenpeace Belgium, micro-cracking in Belgian atomic reactor pressure vessels (RPVs) due to hydrogen flaking could be a global problem -- including in the U.S. -- going undiagnosed, simply because nuclear utilities and government regulators (such as NRC) haven't done, or haven't required, the needed testing. Belgium's nuclear regulatory agency has issued "a statement confirming that the additional tests conducted in 2014 revealed 13,047 cracks in Doel 3 and 3,149 in Tihange 2," as reported in Greenpeace Belgium's press release.

Embrittlement can lead to RPV failure due to pressurized thermal shock (PTS) in pressurized water reactors (PWRs). Beyond Nuclear, in coalition with Don't Waste MI, MI Safe Energy Future, and Nuclear Energy Information Service, has challenged the continued operation of Entergy Nuclear's Palisades atomic reactor in s.w. MI on the Lake Michigan shore, due to its worst embrittled RPV in the U.S. As reported by Greenpeace Belgium, a RPV breach due to PTS could cause a Loss-of-Coolant-Accident (LOCA), core meltdown, containment failure, and catastrophic radioactivity release.

Greenpeace Belgium's experts call for comprehensive testing of all atomic reactors worldwide, a position echoed by Belgium's top nuclear regulator. Belgium's two suspect reactors are currently shut; Greenpeace demands they remain so till the concern is addressed.

Meanwhile, Palisades -- the worst embrittled atomic reactor in the U.S. -- operates at full power. So too do other badly embrittled U.S. atomic reactors, such as Point Beach Unit 2 in WI, also located on the Lake Michigan shore, as is Palisades.

Lake Michigan, wedged between the two worst-embrittled RPVs in the U.S., is a headwaters of the Great Lakes -- 20% of the world's surface fresh water; nearly 90% of North America's surface fresh water; drinking water supply for 40 million people in 8 U.S. states, 2 Canadian provinces, and a large number of Native American First Nations downstream; and lifeblood for one of the world's largest regional economies.

Given the high risks (the consequences to Lake Michigan and the Great Lakes downstream) of a RPV throughwall crack, 10CFR50.61a's increased risks, as compared to 10CFR50.61's, cannot be allowed to stand in NRC regulations, regulatory guides, and supporting technical documents, such as DG-1299 and NUREG-2163.

The Greenpeace Belgium report, and press release, focus on the global risk implications (including in the U.S.) of widespread, sub-surface, micro-cracking revealed under Ultrasonic Testing (UT) at the troubled Doel-2 and Tihange-3 atomic reactors in Belgium. Initially, in 2012-2013, the cracking was thought to be related to defective manufacturing processes at Rotterdam Drydock Company. However, materials scientists now hypothesize that hydrogen flake corrosion cracking is involved -- and are concerned this could impact reactor pressure vessels worldwide (including in the U.S.). They call for precautionary testing of all RPVs

globally, a warning currently being rejected by the nuclear power industry and its regulators internationally -- including in the U.S. by NRC. Such an unacceptable NRC response, under direct questioning by myself, occurred on Feb. 19, 2015, during a day-long NRC meeting about RPV risks which I attended by phone.

Entergy Nuclear Palisades' exhaustively documented RPV embrittlement/PTS risks, in combination with synergistic RPV fabrication or operationally-induced (and worsening with time) flaws -- as from hydrogen flaking -- represent unknown, and very likely unacceptable, risk levels. These synergistic risks are completely unaddressed in 10CFR50.61a, and its associated DG-1299 and NUREG-2163. These synergistic risks must be comprehensively understood, and addressed. DG-1299, NUREG-2163, and 10CFR50.61a do not accomplish this.

Previously undetected, widespread micro-cracking revealed in Belgian RPVs is deeply relevant to the DG-1299/NUREG-2163 proceeding. After all, flaws in the RPV are an essential contributor to PTS though-wall fracture risks (the others being RPV embrittlement, as well as a PTS disaster sequence). Entergy Nuclear at Palisades, and other nuclear utilities with dangerously embrittled RPVs, must comprehensively understand, and address, the number of flaws in RPVs (including micro-cracks), as well as their size, significance, and vulnerability to embrittlement induced PTS risks.

The place to start is UT testing of all RPVs, as carried out at the two troubled Belgian reactors. Top priorities for such testing are at Palisades and Point Beach Unit 2. NRC's DG-1299/NUREG-2163/10CFR50.61a should require this.

NRC Response to BN-9-1

The NRC disagrees with the comment. Various parts of this comment pertain to the operation of specific nuclear power plants and, as such, are outside of the scope of this public comment request. Likewise, other parts of this comment constitute challenges to the provisions of 10 CFR 50.61a. 10 CFR 50.61a was noticed for public comment on October 3, 2007 [72 FR 56275] and August 11, 2008 [73 FR 46557]; it is not the topic of this public comment process. For information, the NRC notes that ultrasonic testing (UT) of reactor pressure vessels is a requirement of 10 CFR 50.61a, as mandated in 10 CFR 50.55a and ASME Code Section XI, as suggested in the last paragraph of this comment. Finally, the portion of the comment, "*These synergistic risks are completely unaddressed in 10CFR50.61a, and its associated DG 1299 and NUREG-2163*" are addressed in the underlying probabilistic fracture mechanics analyses that form the basis for 10 CFR 50.61a. These analyses, which are detailed in reports NUREG-1806 , NUREG-1874 and the supporting documents referenced in each of these NUREGs, featured a comprehensive examination of all credible time-related damage mechanisms for the RPV.

The response to Comment ACE-1-1, which appears beginning on page 3 of this document, also addresses the parts of this comment concerning the recent operational experience in Belgium.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-10-1

Please see the attached file, Official Transcript of Proceedings, Nuclear Regulatory

Commission, Title: Entergy Nuclear Operations, Inc., Palisades Nuclear Plant, Docket Number: 50-255-LA, ASLBP Number: 15-936-03-LA-BD01, Location: Rockville, Maryland, Date: Wednesday, March 25, 2015 (135 pages).

Please accept our (our environmental coalition's -- Beyond Nuclear, Don't Waste MI, MSEF--Shoreline Chapter, NEIS -- as articulated by our attorney, Terry Lodge) numerous challenges and criticisms contained therein, in the context of Entergy Nuclear's July 2014 License Amendment Request for 10CFR50.61a regulatory relief, as public comments in your DG-1299/NUREG-2163 proceeding.

NRC Response to BN-10-1

See NRC response to Comment BN-5-1.

Comment BN-11-1

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-BDO1, 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?

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.

NRC Response to BN-11-1

The NRC disagrees with the comment. The comment mis-characterizes reactor vessel fabrication practices in that it implies that "*…uncontrolled, and unknown, amounts of soft metals, such as copper, manganese, nickel, and phosphorous, as impurities…*" exist in reactor vessel materials. All U.S. nuclear reactor vessels now in operation were fabricated to the requirements of Section III of the ASME Code[1]. Compliance with ASME standards also requires that the materials used in the construction of the facilities meet ASME's procurement, fabrication, inspection, and quality standards. ASME material standards specify limits for the content of all trace chemical elements that can alter the mechanical properties of the steel and are known to enhance irradiation embrittlement, and certification to these standards by Authorized Nuclear Inspectors (ANIs), is required before the materials are placed into service. Therefore, contrary to the statements made in this comment, the chemical content of reactor vessel materials was tightly controlled during the manufacture of those materials. Furthermore, the statistical tests of 10 CFR 50.61a, in conjunction with the reactor vessel surveillance programs required by Appendix H to 10 CFR Part 50, are specifically intended to capture plant-specific embrittlement trends and adjustments.

With respect to the comment that 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…,*" this issue was the subject of a Consumers Power Company license amendment request to the NRC dated August 3, 1983. An NRC safety evaluation approving that request was issued on February 28, 1984 (ADAMS Accession No. ML020800206). This matter was brought forth to the ASLB Board in a declaration by Mr. Gunderson dated December 1, 2014 (ADAMS Accession No. ML14335A806). The staff replied to Mr. Gunderson's challenge in their Answer dated January 12, 2015 (ADAMS Accession No. ML15012A611) where they noted that withdrawal and testing of Capsule A-60 was unnecessary. The ASLB already addressed this matter in their Memorandum and Order dated May 8, 2015 (ADAMS Accession No. ML15128A489) at pages 30-34, and the Board ruled that, although the Petitioners demonstrated standing to intervene, they did not put forward an admissible contention. Therefore, this portion of the comment was previously addressed and is further not within the scope of the request for comments on DG-1299 and NUREG-2163 because it does not address these documents.

The remainder of this comment contains no further specific issues associated with DG-1299 or NUREG-216. As such those portions of this comment lie beyond the scope of this request for public comment so they are not addressed here.

No changes were made to the DG or NUREG in response to this comment.

[1] According to the Appendix A of NRC Information Digest NUREG-1350, Volume 27 (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1350/) the oldest pressurized water reactor still operating in the United States today is located at the Robert E. Ginna nuclear power plant in Ontario, New York. The NRC issued a construction permit for the Ginna plant on April 25, 1966. The original edition of Section III of the ASME Code was issued in 1963, three years before the Ginna construction permit was issued. The 1963 Edition of the ASME Code required N-stamps for fabrication of nuclear vessels (https://www.asme.org/shop/certification-and-accreditation/nuclear-component-certification); N-stamps certify that the vessel was fabricated to the requirements of Section III.

Comment BN-12-1

Please see the attached file:

March 17, 2006: Petitioners Notice of Appeal from ASLB Denial of Hearing, and Supporting Brief, submitted to the U.S. NRC ASLB, by attorney Terry Lodge, on behalf of Don't Waste Michigan and NIRS, in opposition to Palisades' 20-year license extension (Appeal of dismissal of Contention No. 1, The license renewal application is untimely and incomplete for failure to address the continuing crisis of embrittlement, specifically pages 3 to 9, as well as portions of the conclusion relevant to PTS risks/RPV embrittlement).

Please accept the numerous challenges and criticisms contained within this document, in the current context of Entergy Nuclear's July 2014 License Amendment Request for 10CFR50.61a regulatory relief, as public comments in your DG-1299 and NUREG-2163 proceeding. At the time, in 2005-2006, 10CFR50.61 was the ruling regulatory regime. The concerns raised by attorney Terry Lodge on behalf of intervening groups NIRS and Don't Waste MI at the time, are all the more poignant now, that an even less conservative alternate fracture toughness rule (10CFR50.61a) is the context for this proceeding on DG-1299 and NUREG-2163.

NRC Response to BN-12-1

The NRC disagrees with the comment. The referenced attachment concerns a license amendment request for the Palisades Nuclear Power Plant for a 20-year license extension. This petition was the subject of hearings before the Atomic Safety Licensing Board. The matter was addressed by the Board in their Order issued on June 23, 2006 (ADAMS Accession No. ML061740303) where the ASLB Panel ordered that their previous decision to deny the Petitioner's hearing and supporting brief was affirmed. The Petitioner's contentions were previously addressed by the ASLB Panel in their March 7, 2006 Memorandum and Order (ADAMS Accession No. ML060660560).

Furthermore, the referenced attachment does not directly pertain to any provision of DG-1299 or NUREG-2163. As such this comment lies beyond the scope of this request for public comment so it is not addressed here.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-15-1

Please see the attached files:

1. *August 8, 2005: Request for Hearing and Petition to Intervene, submitted to the U.S. NRC ASLB, by attorney Terry Lodge, on behalf of Don't Waste Michigan and NIRS, in opposition to Palisades' 20-year license extension (specifically, the first contention, beginning on page 4, regarding "The license renewal application is untimely and incomplete for failure to address the continuing crisis of embrittlement").*
2. *September 16, 2005: Petitioners Combined Reply to NRC Staff and Nuclear Management Company Answers, submitted to the U.S. NRC ASLB, by attorney Terry Lodge, on behalf of Don't Waste Michigan and NIRS, in opposition to Palisades' 20-year license extension (pages 2 to 23 are regarding Contention 1, The license renewal*

application is untimely and incomplete for failure to address the continuing crisis of embrittlement).

3. *Petitioners' Appendix of Evidence (129 pages), which accompanied its September 16, 2005 Reply.*
4. *November 3, 2005: Transcript of oral argument pre-hearing before the NRC ASLBP, re: 20-year license extension for Palisades. The hearing was held in South Haven, Michigan. (See, specifically, the portions pertaining to PTS risks, including pages 34-80 (pages 17-63 of 206 on PDF counter), and following, as articulated by attorney Terry Lodge on behalf of intervening groups NIRS and Don't Waste MI.)*

Please accept the numerous challenges and criticisms contained within these documents, in the current context of Entergy Nuclear's July 2014 License Amendment Request for 10CFR50.61a regulatory relief, as public comments in your DG-1299 and NUREG-2163 proceeding. At the time, in 2005, 10CFR50.61 was the ruling regulatory regime. The concerns raised by attorney Terry Lodge on behalf of intervening groups NIRS and Don't Waste MI at the time, are all the more poignant now, that an even less conservative alternate fracture toughness rule (10CFR50.61a) is the context for this proceeding on DG-1299 and NUREG-2163.

NRC Response to BN-15-1

The NRC disagrees with the comment. NRC is familiar with the four files attached to the comment, as follows:

1. August 8, 2005 Request for Hearing and Petition to Intervene submitted to the ASLB Panel containing eleven specific contentions against Entergy's application for a 20-year license extension for the Palisades plant (ADAMS Accession No. ML052940221).
2. September 16, 2005 Petitioners Combined Reply to NRC Staff and Nuclear Management Company Answers, submitted to the U.S. NRC ASLB in response to the NRC staff reply opposing the request for hearing and petition to intervene (ADAMS Accession No. ML052770083).
3. Petitioners' 129-page Appendix of Evidence containing 23 exhibits that accompanied their September 16, 2005 Petitioners Combined Reply (ADAMS Accession No. ML052720270).
4. November 3, 2005 Transcript of the ASLB Panel Limited Appearance Public Hearing in South Haven, MI regarding the 20-year license extension for Palisades (ADAMS Accession No. ML053130107).

The attachments to this comment concern a license amendment request for the Palisades Nuclear Power Plant for a 20-year license extension. This petition was the subject of hearings before the Atomic Safety Licensing Board. The matter was addressed by the Board in their Order issued on March 7, 2006 (ADAMS Accession No. ML060660560) where the ASLB Panel concluded that, although the Petitioners established standing to participate in that proceeding, they did not show a good cause for the ASLB Panel not to rule on their contentions at that time. In addition, the Petitioners' objections and motion were denied because they did not proffer any admissible contention, nor did they establish grounds for granting a hearing.

Furthermore, the referenced attachments do not directly pertain to any provision of DG-1299 or NUREG-2163. As such this comment lies beyond the scope of this request for public

comment so it is not addressed here.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-16-1

Please note that in his first article [http://www.ap.org/company/awards/part-i-aging-nukes] in his four-part series in 2011 entitled "Aging Nukes" (http://www.ap.org/company/awards/aging-nukes), AP investigate reporter Jeff Donn listed weakening of pressurized thermal shock/brittle fracture safety standards as a top example of complicity by NRC in accommodating age-degraded reactors, so they could keep operating.

I would also note that the reporter interviewed the NRC staff whistleblower, Demetrios Basdekas, who brought PTS risks to public light over 30 years ago. Basdekas, and the article, debunked NRC's supposed basis for 10CFR50.61a, DG-1299, and NUREG-2163 in the first place - supposed over-conservatism in 10CFR50.61. That's what NRC and industry always say, essentially!

NRC Response to BN-16-1

The NRC disagrees with the comment. The NRC responded to the 2011 article in a letter dated June 21, 2011, "NRC Ensures Public Safety through Rigorous Oversight of Nuclear Power Plant Safety Standards" (available at ADAMS Accession No. ML11174A232). In that response, the NRC noted their disagreement with many of the observations and conclusions made in the article, described how the NRC sets appropriate technical requirements using impartial professional standards, expertise and analysis, identified that the NRC has inspectors stationed at every nuclear power plant in the country who inspect plants every day, and reiterated how the NRC enforces our requirements to ensure the public remains safe.

Because the article raised in this comment was previously addressed by the NRC, and because this comment contains no further specific issues associated with DG-1299 or NUREG-2163, no changes were made to the DG or NUREG in response to this comment.

Comment ANON-13-1

As your NUREG implies, neutrons are not the only thing causing embrittlement of nuclear reactor pressure vessels (RPV) and they are aging faster than the models predicted and you must test all RPVs. If not, you are guilty of criminal negligence: "If they are statistically significant, Type A, B, and D deviations all give rise to concerns that the embrittlement trends predicted by the ETC may produce non-conservative estimates of the embrittlement experienced by materials used to construct the RPV that is being evaluated... Type C deviations, if they are statistically significant, suggest that the surveillance program for the material in question may not provide a reliable indication of embrittlement trends for that material. Because Appendix H to 10 CFR 50 requires the performance of surveillance on the limiting (meaning most irradiation-sensitive) materials used to construct the RPV beltline, the existence of a Type C deviation is important from a regulatory viewpoint," (p. 27). The sentence continues with NRC typical nonsense: "but not in the context of indicating a potential

nonconservatism in the predictions of the T30 ETC adopted in the Alternate PTS Rule. For this reason, statistical procedures to detect Type C deviations were not included in the Alternate PTS Rule..." (p. 27).

This NUREG is an F - in Methodology and Stats. To start being statistically significant you must have at least 50 of whatever you are studying and for testing go up to 120 or more. This is true in both science and social science. The entire population of US reactors is 100 and 2/3rds are PWR. Thus, you have to test all of them. You cannot test 2 or 3 and extrapolate to the others, as you seem to be doing. It's like saying that your neighbor's vote will tell you the result of an election. And, when you do your little tests of whatever you are testing – which isn't really clear and which seems to be strictly theoretical - whatever you are testing you need 50 to 120 or more data points for each reactor and not hovering around 3 to 8 and within a zero to 24 range. This is not statistically significant even if you are measuring something which matters. Furthermore the 50 to 120 data points themselves must have meaning and significance and not be some weird theory like stock derivatives.

So pitiful is the state of these nuclear reactors, that even your incomplete and lamentable stats show that they are aging faster than the models predict. They must be shut down now.

Stop cheating with statistics & trying to kill everyone & everything. This is my love letter for the trees, plants, animals & the unborn children. Shame on you.

NRC Response to ANON-13-1

The NRC disagrees with the comment. The NRC understands that, in this part of the ANON-13 comment, two objections are raised on the requirements set forth in the Alternate PTS Rule (10 CFR 50.61a): (1) the limited amount of material testing required, and (2) the appropriateness of the statistical tests used to compare plant-specific data to general trends.

Regarding the first objection, the NRC notes that the requirements for the surveillance program, including the number of capsules that must be tested and the schedule for their testing, is established separately under 10 CFR Part 50, Appendix H. Neither 10 CFR 50.61a nor DG-1299 establishes any requirements for surveillance testing beyond those of 10 CFR Part 50, Appendix H. As such, the quantity of plant specific data available to analyze is established in other regulations and is not a topic pertinent to this request for public comment.

Regarding the second objection, the NRC notes that the statistical tests in 10 CFR 50.61a, which are described in greater detail in DG-1299, employ standard procedures for comparing trends quantified by plant specific data to the trends predicted by the embrittlement trend curve (ETC) in 10 CFR 50.61a. In these statistical tests, the number of plant specific data is explicitly accounted for by factors in the statistical evaluation that vary depending on the number of plant specific data available for assessment (see Tables 5 and 6 of 10 CFR 50.61a). These factors account for the data set size so that the statistical evaluations can appropriately discern when the plant specific data differ significantly from the predictions of the ETC. In the event that the plant specific data show a trend towards more embrittlement than predicted by the ETC, adjustments are required to ensure that the embrittlement behavior is bounded. In this manner, the statistical evaluations required by 10 CFR 50.61a ensure that plant-specific embrittlement behavior is accounted for appropriately.

No changes were made to the DG or NUREG in response to this comment.

Comment ANON-13-2

Dr. Digby McDonald, US corrosion expert, who has done research for the DOE, similarly remarked that the Belgium nuclear reactors had more flaws than the models had predicted. He says all reactors in the world need to be properly tested with ultrasound. You cannot depend on theoretical models. All RPV must be tested on a sensitive setting. Any of the material test samples within the reactor must be removed and tested now and not later.

Not only do neutrons cause embrittlement, which can lead to reactor pressure vessel failure, hydrogen attack can also lead to nuclear reactor pressure failures, as can corrosion. They all work together synergistically to weaken the nuclear reactor pressure vessel. They must be evaluated together on each unique nuclear reactor pressure vessel.

This NUREG is fraudulent because whereas pressurized thermal shock of PWRs is more likely to cause sudden RPV failure, it can happen at any time and also with Boiling Water reactors. Pressurized reactors are more at risk, but all are at risk. It's not just a shift to cold temperatures which can cause it. The petroleum industry shows that hydrogen attack can happen under heated conditions. Also, neutron bombardment shifts the ductile-brittle transition upwards so that they can fail even under normal operating conditions. Davis Besse's RPV showed the dangers of corrosion.

NRC Response to ANON-13-2

The NRC disagrees with the comment. The comment mentions "…*Belgium nuclear reactors*…" and "…*hydrogen attack*…" and "…*Davis Besse's RPV showed the dangers of corrosion*…" Issues of hydrogen flaking (such as that experienced in Belgium) and corrosion (such as that experienced by Davis Besse) are not the subject of the guidance contained in DG-1299 or NUREG-2163. Inspections performed in accordance with Section XI of the ASME Code are aimed at finding possible degradation caused by these and other aging mechanisms. If detected, Section XI of the ASME Code, which is mandated for use by 10 CFR 50.55a, requires an appropriate disposition of the findings, which may include evaluation, repair or replacement of the affected component. Depending on the severity of the findings, NRC approval prior to plant restart may be required.

No changes were made to the DG or NUREG in response to this comment.

Comment ANON-13-3

As the Belgium regulated learned, the ultrasound must be placed in the most sensitive mode or defects can be missed. To be conservative they lumped smaller defects together, assuming that they could merge to one. Your NUREG is doing the opposite and de-clumping defects and falsely assuming that the ultrasound over-estimates. Furthermore, you can NOT assume that thickness matters and that cracks inside do not matter, as you do. The neutrons can most certainly impact the middle of the material and the hydrogen may as well. With brittle fracture thickness matters much less than material quality. You must test the entire thickness.

Additionally you must test each nuclear RPV thoroughly or shut them down. Each nuclear

reactor is unique in manufacture, construction, and operating history. You cannot extrapolate from a few to all, except to conclude that ALL must be tested!

NRC Response to ANON-13-3

The NRC disagrees with this comment. The comment seems to be referring to the inspection aspects of the DG and NUREG, in that it refers to "…*properly tested with ultrasound*…" and "…*the ultrasound must be placed in the most sensitive mode or defects can be missed*…" and "…*you must test each nuclear RPV thoroughly*…" Section XI of the ASME Code requires nondestructive volumetric examination of all reactor vessel welds on a periodic basis. 10 CFR 50.55a, "Codes and Standards," mandates that all plants implement the provisions of Section XI of the ASME Code. 10 CFR 50.55a also mandates the use of Appendix VIII to Section XI of the ASME Code, which provides rigorous procedures and qualification requirements for ASME-mandated examinations to ensure that sufficient sensitivity is used in the conduct of the examinations. Furthermore, Appendix VIII Supplements 4 and 6, as referenced by DG-1299 and NUREG-2163, require interrogation of the full volumetric wall thickness for all reactor vessel welds. Therefore, independent of DG-1299, it is a requirement for all U.S. plants, both BWRs and PWRs alike, to examine all reactor vessel welds on a periodic basis.

10 CFR 50.61a mandates the use of Appendix VIII examination results in the evaluation of pressurized thermal shock (PTS). Only the examination results for the inner one inch or ten percent of the wall thickness are considered in the PTS evaluation required by 10 CFR 50.61a because, as explained in NUREG-1806 (see http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1806/), only flaws in that portion of the wall thickness are subject to tensile loading and, thereby, can contribute to PTS failure risk. Examination and, if necessary, flaw evaluation of detected flaws are still required for the remaining wall thickness in accordance with Section XI as mandated by 10 CFR 50.55a.

No changes were made to the DG or NUREG in response to this comment.

Comment DWM-14-1

The U.S. Nuclear Regulatory Commission (NRC) proposed utilization of regulatory guide (DG), DG-1299, Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule and accompanying draft NUREG-2163, Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule is a misuse and abuse of methodological principles. Rather than the NRCs reliance on apples, oranges, bananas and watermelon as sister plant comparisons the NRC might better compare lemons to lemons. This entire fabrication and rationalization is for one nuclear plant and that plant is a lemon. The reliance of sister plant data to justify continued operation of the Palisades nuclear plant, known to have exceeded embrittlement as early as 1981 approaches aiding and abetting a criminal enterprise. Pull the capsules and look at real data, reliance on proxy data is absolutely fudging scientific inquiry.

The proposed DG-1299 provides slight of hand pseudo methodologies and instructions as to how to subvert inquiry into fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor (PWR) reactor pressure vessels (RPVs). This thinly veneered draft NUREG-2163 attempts to provide rationalization and justifications for the utilization of the technical basis for DG-1299 which is intended to subvert

tangible, proven methods of inquiry into embrittlement. This is about covering the NRCs ass when things go south, as in catastrophic pressure vessel failure. Pull the capsules and look at the real data. Please stop the BS on excessive conservatism recapture with this fudged methodology.

The reliance on DG-1299 provides a scheme by which the NRC will ignore prima facie evidence and provides a fig leaf for the NRC to once again capitulate to production over public safety.

The NRCs consideration of this fabricated and strained methodology as an acceptable alternative fracture toughness requirement for protection against PTS events for PWR RPVs does not pass the smell test. The Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events predicts and explains nothing other than the extraordinary measures that the NRC and Entergy are willing to engage to dupe the public into acceptance of running into the ground the dilapidated Palisades limited liability (LLC) nuclear power plant. Once again production over public safety.

This contrived and strained process is all about Palisades and this scheme was cooked up in the late 1990s to keep Palisades running and subvert well established guidelines.

Time to pull the plug before catastrophic failure which the NRC and Entergy seem hell bent on delivering.

NRC Response to DWM-14-1

The NRC disagrees with this comment. The NRC understands that, in this comment, two objections are raised on the requirements of the Alternate PTS Rule (10 CFR 50.61a): (1) the lack of a requirement to test additional surveillance capsules, and (2) the requirement to assess data from similar plants (so-called "sister" plants). Regarding the first objection, the NRC notes that the requirements of the surveillance program are administered separately under 10 CFR Part 50, Appendix H. Regarding both objections, the NRC notes that neither DG-1299 nor NUREG-2163 establishes any new requirements, changes any requirements, or establishes any new policies with regards to testing surveillance capsules. As such, these comments do not directly pertain to any provision of DG-1299 or NUREG-2163.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-17-1

Please see attached file:

Spring 2006: Consumers Energy power point presentation to the Michigan Public Service Commission, highlighting "Reactor vessel embrittlement concerns" at Palisades.

Please note that the DG-1299 and NUREG-2163 proceedings are part and parcel of a long line of broken promises made at Palisades, the worst embrittled RPV in the U.S., and the only one currently applying for 10CFR50.61a regulatory relief.

NRC Response to BN-17-1

The NRC disagrees with this comment. With regard to page 2 of the Consumers Energy

presentation that was included with the comment, the NRC does not see any specific content in that presentation related to DG-1299 or NUREG-2163, nor does the commenter provide any specific comments related to DG-1299 or NUREG-2163 in their reference to that presentation. The NRC notes that the provisions of 10 CFR 50.61a apply to *all* operating pressurized water reactors; they are not specific to the Palisades Nuclear Plant.

Because this comment contains no specific issues associated with DG-1299 or NUREG-2163, no changes were made to the DG or NUREG in response to this comment.

Comment BN-17-2

In addition to such broken promises as the 1970 ACRS assurance to AEC that should RPV embrittlement reach a dangerous enough point, annealing would take place, there is this spring 2006 promise by Palisades' previous owner, Consumers Energy, that should the Michigan Public Service Commission approve the sale of Palisades to Entergy, the much larger new owner, with vastly more nuclear power experience as well as economy of scale and expertise, would solve the problem of "Reactor vessel embrittlement concerns."

Far from fulfilling that false promise, Entergy has instead applied for an LAR for regulatory relief under 10CFR50.61a.

DG-1299 and NUREG-2163 should be withdrawn. They are unacceptable on their face, for being part and parcel of a long history of broken promises, and weakening of safety regulations, at Palisades in particular.

But equally objectionable is the fact that bad precedents set at Palisades, as under the DG-1299/NUREG-2163/10CFR50.61a proposed, weakened regulatory regime, would then be applied at other dangerously embrittled atomic RPVs across the U.S. As mentioned in our previous comments in this proceeding, this could include Point Beach Unit 2 in WI, on the other side of Lake Michigan.

However, to the best of our knowledge, it would appear that Point Beach Unit 2 has missed its three-years-in-advance deadline to make LAR application for 10CFR50.61a regulatory relief. For, in March/April 2013, specifically, orally during a Palisades PTS risk Webinar conducted by NRC Region 3 staff on March 19, 2013, as documented in an April 18, 2013 NRC document (point #4, page 2, or page 5 of 15 on PDF counter – see attached here), NRC listed Point Beach Unit 2 as violating 10CFR50.61 screening criteria by 2017, just like Palisades.

Regarding this, our comment, in this DG-1299 and NUREG-2163 proceeding, is that any atomic reactor that misses such a deadline, must simply permanently shutdown when the 10CFR50.61 screening criteria are surpassed. After all, as environmental intervenors were chastised in early 2006 by the ASLBP in the Palisades' license extension proceeding, when the ASLBP rejected the intervention petition, NRC's regulations are "strict by design" -- or at least they should be, when it comes to enforcing PTS risk standards at dangerously embrittled RPVs.

Along those same lines, we, and others before us, have long called for Palisades' permanent shutdown for surpassing 10CFR50.61 screening criteria as early as 1995.

NRC Response to BN-17-2

The NRC disagrees with this comment. With regard to the "*1970 ACRS assurance to AEC*" mentioned in the comment, the NRC assumes that the commenter is referring to the second attachment provided in Comment BN-11-1 (ACRS Memo from ACRS Chairman Joseph M. Hendrie to AEC Chairman Glenn T. Seaborg, "Report on Palisades Plant," January 27, 1970). In that attachment, the ACRS briefly described a material surveillance program for the Palisades plant that is consistent with current federal regulations, as specified in 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." As documented in the latest staff Safety Evaluation (SE) pertaining to the Appendix H surveillance program (ADAMS Accession No, ML071640310), *"the proposed surveillance capsule withdrawal schedule for the Palisades Nuclear Plant is in accordance with Appendix H to 10 CFR Part 50".* This SE evaluated the suitability of the surveillance program *"to accommodate the 60-year licensing period."* Furthermore, provisions for thermal annealing of the reactor vessel are contained in 10 CFR 50.66, "Requirements for Thermal Annealing of the Reactor Pressure Vessel." The provisions of 10 CFR 50.66 are optionally available for use by the licensee to recover fracture toughness of the reactor pressure vessel material in two cases: (1) if the limiting RT_{PTS} value of the facility is projected to exceed the screening criteria of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and the requirements of 10 CFR 50.61 cannot be satisfied, or (2) if the limiting RT_{MAX-X} value of the facility is projected to exceed the screening criteria of 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and the requirements of 10 CFR 50.61a cannot be satisfied. These thermal annealing provisions remain available for use at Palisades in the event relevant PTS screening criteria and associated requirements cannot be satisfied. Neither Consumers Power, nor Entergy, have committed to these provisions for the Palisades plant.

In July 2014, Entergy submitted a license amendment request (LAR) to implement the provisions of 10 CFR 50.61a for the Palisades plant (ADAMS Accession No. ML14211A520). The comment is incorrect that this request was for regulatory relief. The provisions of 10 CFR 50.61a are alternative requirements to those of 10 CFR 50.61. As such, regulatory relief is not applicable; the LAR requested permission to use the alternate requirements of 10 CFR 50.61a in place of their previously accepted use of 10 CFR 50.61.

On January 15, 2013, NextEra Energy submitted a LAR for the Point Beach plant requesting, in part, an exemption from portions of 10 CFR 50.61 (ADAMS Accession No. ML13016A028). The LAR requested exemption from 10 CFR 50.61 to allow the use of fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the Point Beach reactor vessel beltline regions, as contained in Topical Report BAW-2308, Revision 1-A, "Initial RT_{NDT} of Linde 80 Weld Materials." The NRC's review of the Point Beach LAR was still underway during the PTS Webinar conducted by NRC Region 3 staff on March 19, 2013. The NRC granted the exemption request for Point Beach on June 30, 2014 (ADAMS Accession No. ML14126A596). The staff concluded that the use of the BAW-2308, Revision 1-A methodology will ensure the PTS evaluation developed for both of the Point Beach reactor vessels under the provisions of 10 CFR 50.61 will continue to be based on an adequately conservative estimate of RPV material properties and ensure the RPVs will be protected from failure during a PTS event. As a result of this request, the requirements in 10 CFR 50.61 remain satisfied for Point Beach, so a submittal to use the alternate requirements of 10 CFR 50.61a was not necessary. Therefore, the 3-year

deadline required by 10 CFR 50.61a was not missed for Point Beach.

Finally, the comment that "*any atomic reactor that misses such a deadline, must simply permanently shutdown when the 10CFR50.61 screening criteria are surpassed*" mis-characterizes the legal requirements of 10 CFR 50.61. The screening criteria of 10 CFR 50.61 were established as a threshold beyond which additional evidence is needed to demonstrate the continued operating safety of the reactor. The screening limits never were intended to be, and should not be construed as, absolute limits on reactor operation. To address this part of the comment, rulemaking would need to be undertaken to change the provisions of 10 CFR 50.61.

No changes were made to the DG or NUREG in response to this comment.

Comment BN-17-3

In fact, Michael Keegan of Coalition for a Nuclear-Free Great Lakes, as well as Don't Waste MI, called for Palisades' permanent shutdown in 1993. See his attached document "Pressurized Thermal Shock Potential at Palisades," dated July 8, 1993. In it, he pointed out that Palisades' first violated RPV embrittlement safety standards, as then extant, as early as 1981, just ten short years into its operations. In fact, it is very likely that Palisades' embrittlement problems contributed significantly to the promulgation of 10CFR50.61 in the first place, in the mid-1980s, as it has long been among the worst, and even the very worst, embrittled RPV in the entire country.

This long history, spanning several decades, of Palisades' dangerously embrittled RPV being accommodated by NRC, through the weakening of safety standards -- as is now proposed under 10CFR50.61a, DG-1299, and NUREG-2163, must stop. DG-1299 and NUREG-2163 must be withdrawn, just as Entergy's LAR for 10CFR50.61a regulatory relief must be rejected.

NRC Response to BN-17-3

The NRC disagrees with this comment. The NRC previously addressed these same comments appearing in the July 8, 1993 publication, "Pressurized Thermal Shock Potential at Palisades," that relate to potential violations of PTS safety standards in their summary of the March 19, 2013 public meeting webinar (see Question 6 of Enclosure 2, ADAMS Accession No. ML13108A336).

No changes were made to the DG or NUREG in response to this comment.