



**UNITED STATES  
NUCLEAR REGULATORY COMMISSION**  
REGION III  
2443 WARRENVILLE ROAD, SUITE 210  
LISLE, IL 60532-4352

April 18, 2013

LICENSEE: Entergy Nuclear Operations, Inc.  
FACILITY: Palisades Nuclear Plant  
SUBJECT: SUMMARY OF THE MARCH 19, 2013, PUBLIC MEETING WEBINAR  
REGARDING PALISADES NUCLEAR PLANT

On March 19, 2013, the U.S. Nuclear Regulatory Commission (NRC) held a two part Public Meeting webinar to discuss NRC's perspectives on pressurized thermal shock (PTS). During the first part of the meeting, the NRC staff presented an overall discussion regarding the basics of embrittlement and PTS, and the regulatory requirements that apply to PTS. Enclosure 1 is a list of attendees at the meeting. Copies of the slides used by the NRC staff during the meeting can be accessed through the NRC's Agency wide Document Access and Management System: ADAMS (ML13077A156).

The NRC staff stated in the opening remarks that the second part of the meeting was geared towards answering follow up questions from the public about PTS. There were 118 meeting participants that had the opportunity to submit questions to the NRC staff about PTS issues through the Webinar process. There is one clarification for information provided during the Webinar. During the presentation it was stated that there were two capsules left to determine properties of neutron irradiation based on the Safety Evaluation Report for license renewal of the Palisades Plant. However, since license renewal approval, Palisades requested and the NRC approved a schedule change which left an additional capsule in the reactor pressure vessel (RPV) which would have been removed in 2007. Therefore there are three capsules in the RPV which can be used to determine properties of neutron irradiation. One capsule is scheduled to be removed during the period of extended operation, and this is tentatively scheduled around 2019. In addition to the capsules used to determine properties of neutron irradiation, one thermal capsule is also in the vessel which can measure thermal exposure effects on the metal, and is available for future use. So there are currently a total of four capsules in the RPV.

In addition to answering questions from members of the public on March 19, NRC representatives agreed to provide an answer to technical questions regarding the topic of PTS that were submitted during the meeting, but were not answered during the allocated meeting time. The answers to these questions and the follow-up to two questions answered during the webinar are included in this meeting summary (Enclosure 2). Availability of a recording of the webinar will be addressed by separate correspondence.

Sincerely,

***/RA/***

John B. Giessner, Chief  
Branch 4  
Division of Reactor Projects

Docket Nos. 50-255 and 72-007  
License No. DPR-20

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**PUBLIC MEETING PRINCIPAL ATTENDEES**

**March 19, 2013**

NRC Attendees

C. Pederson, Deputy Regional Administrator, RIII

J. Giessner, Chief, Division of Reactor Projects, Branch 4

M. Kirk, Senior Materials Engineer, Office of Nuclear Regulatory Research

M. Holmberg, Senior Reactor Inspector, RIII

**Questions for the NRC Meeting on March 19, 2013**

**1. Does the public need to fear Palisades continued operation?**

No, the NRC's oversight of Palisades continues to show that the plant is operating safely. If at any point the NRC deemed Palisades to be unsafe, the NRC would take action to shut down the plant

**2. On September 14, 2011, during the loss of power to half the control room at Palisades, the emergency core cooling system was inadvertently activated. If it had operated as instructed, albeit inadvertently, could the pressurized thermal shock on the 100 percent power level and heat level Palisades RPV have fractured under the sudden temperature plunge, coupled with the high pressure level?**

No, had both systems actuated and injected as designed, a pressurized thermal shock would not have occurred and the vessel would not have fractured. During the September 14, 2011, event the 'A' train of emergency core cooling system was activated due to a Safety Injection Actuation Signal being present. The two pumps that comprise the 'A' train (the 'A' low pressure safety injection and 'A' high pressure safety injection pumps) did not inject into the vessel due to the primary coolant system (PCS) pressure being higher than the pumps shut off head. Even in the event of lowering PCS pressure due to a loss of coolant accident (LOCA), if an additional fault occurred, the possibility of developing a crack through the vessel wall as a result of PTS would be extremely low.

Additional information on this event is located in NRC Inspection Report 05000255/2011014 (ML113330802).

**3. Given the badly embrittled status of the Palisades RPV, might this not lead Palisades control room operators and senior management hesitating before activating the emergency core cooling system, for fear of fracturing the RPV? Might this not significantly increase the risks of an overheating accident, and even a meltdown?**

As stated during the webinar, Palisades currently remains compliant with the PTS rule contained in 10 CFR 50.61. By being compliant with this rule the probability of developing a crack through the vessel wall as a result of PTS remains extremely low.

Regarding operator actions, the NRC requires, through the site's Technical Specifications, that the emergency core cooling systems be able to operate automatically if called upon during an accident. In the case that existing conditions merit initiation of the emergency core cooling systems (i.e. low pressurizer pressure) the emergency operating procedures instruct the operators to start the safety pumps, if not already running. Operators are highly trained individuals that are licensed by the NRC to respond to such events. In the case of an event they will follow their emergency procedures.

**4. Which are the other most embrittled plants in the U.S.? How many PWRs will reach their screening criteria in the next 10 years?**

The NRC currently estimates that the following plants will exceed the PTS screening criteria of 10 CFR 50.61 during their 20-year period of operation beyond their original 40 year licenses. Updated fluence calculations, capacity factors changes, power uprate, new surveillance data, and improved material property information (i.e., the use of direct rather than correlative measurements of the vessel material's resistance to fracture) can change these estimates. For example, Point Beach has made a recent licensing submittal that seeks to use improved material property information to re-evaluate the level of embrittlement in the vessel. If approved, it is estimated that Point Beach would not exceed the screening criteria of 10 CFR 50.61 during their 20-year license extension period.

1. Point Beach 2 (2017)
2. Palisades (2017)
3. Diablo Canyon 1 (2033)
4. Indian Point 3 (2025)
5. Beaver Valley 1 (2033)

Another method by which nuclear power plants that are projected to exceed the screening criteria of 10 CFR 50.61 may justify their continued safe operation is to prepare a submittal following the requirements of the alternative PTS rule, 10 CFR 50.61a. Such a submittal would employ improved screening criteria that are based on updated and more accurate PTS analyses that were performed by the staff over a 10 year period. To use these improved screening criteria, licensees would need to provide the NRC with evidence that key assumptions regarding embrittlement and flaws that underlie the staff's PTS analysis are satisfied by the nuclear power plant. To date, the licensees for Beaver Valley Unit 1, Palisades, and Diablo Canyon Unit 1 have expressed their intention to submit updated PTS evaluations using 10 CFR 50.61a.

**5. I would like to see the calculations supporting the statement that Palisades is the most brittle**

During the March 19 webinar, it was stated that Palisades is "*one of the most embrittled plants,*" not "*the most brittle.*" One example of these calculations can be found in NUREG-1874 "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (<http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1874/>). Tables 3.3 and 3.4 of this document provide calculations of embrittlement levels made in accordance with 10 CFR 50.61a. These calculations show that after 48 effective full-power years (EFPY) of operation Palisades is the fourth closest plant to the 10 CFR 50.61a screening limits. Similar calculations provided in a different document (Table 1 in ML070570141) show Palisades to be the third closest plant to the 10 CFR 50.61 screening limits. The specific ordering of a plant relative to other plants can change over time as new information becomes available, and differs slightly between 10 CFR 50.61 and 10 CFR 50.61a because of differences in the estimation procedures used in the different rules.

It should be noted that the lists provided in these two documents compare the estimated level of embrittlement in operating reactors for identical operational durations (e.g., 48 EFPY in the case of the first document). Because different reactors began producing power on different dates these comparisons are not the same as a comparison made at a fixed date. Finally, these lists show that the estimated magnitude of embrittlement is quite similar between the leading plants, making the distinction between “the most embrittled plant” and, for example, “the fourth most embrittled plant” insignificant.

**6. My question is, didn't Palisades first violate NRC's PTS safety standards 10 short years into its operation, by 1981? This was documented in the following document: July 8, 1983: "Pressurized Thermal Shock Potential at Palisades: History of Embrittlement of Reactor Pressure Vessels in Pressurized Water Reactors," prepared by Michael J. Keegan, Coalition for a Nuclear Free Great Lakes, Monroe, Michigan (re-published August 3, 2005).**

In the referenced 1983 document by M. Keegan (Coalition for a Nuclear Free Great Lakes) it is stated that:

*"Embrittlement at Palisades in 1981 was reported to occur at temperatures of between 190 and 220 degrees F. As noted earlier the NRC had originally set reference temperature for nil ductility transition ( $RT_{NDT}$ ) at 200 degrees F. As early as 1981 Palisades had exceeded these original  $RT_{NDT}$  limits."*

Note: The  $RT_{NDT}$  term refers to a metric that the NRC uses to quantitatively assess brittleness and can roughly be described as the temperature below which the material transitions from ductile to brittle behavior.

These statements are not accurate in several respects. First and foremost, Palisades did not violate the NRC's PTS safety standards in 1981 since the NRC did not have any regulations pertaining to PTS until June 26, 1984, when 10 CFR 50.61 was first promulgated. The  $RT_{NDT}$  limit of 200 °F incorrectly attributed to PTS in the article appeared in Regulatory Guide 1.99 "Radiation Embrittlement of Reactor Vessel Materials", Revision 1, which was adopted in 1977. This document states:

*"For new plants, the reactor vessel beltline materials should have the content of residual elements such as copper, phosphorus, sulfur, and vanadium controlled to low levels. The levels should be such that the predicted adjusted reference temperature at the  $\frac{1}{4} T$  position in the vessel wall at end of life is less than 200 °F. [These] recommendations ... will be issued in evaluating construction permits docketed on or after June 1, 1977."*

Regulatory Guides do not contain requirements, only recommendations. This recommendation amounted to good practice guidance that new plants should limit copper content in their reactor vessels which, by 1977, was known to promote embrittlement. In any event, this recommendation did not apply to the Palisades plant which received its construction permit on March 14, 1967.

In conclusion, there was no violation of NRC requirements concerning PTS at Palisades. Had Palisades ever violated PTS requirements the NRC would have shut down the plant. The plant is operating safely in compliance with 10 CFR 50.61.

**In addition, the Associated Press's Jeff Donn pointed to NRC's weakening of PTS safety regulations as his top example of NRC weakening safety regulations in order to allow dangerously degraded old reactors to continue operating despite the worsening breakdown phase risks, in his four part series "Aging Nukes," dated June 2011**

As far as weakening NRC safety regulations by approving the alternate rule, 10 CFR 50.61a: as was mentioned during the webinar, this alternate rule is justified by an improved state of both theoretical and practical knowledge, more accurate models, and model validation. There was no weakening of regulations. These developments were made with great deliberation over the 10 year period preceding adoption of 10 CFR 50.61a in 2010. Moreover, the new rule was reviewed extensively, and approved, by the NRC's Advisory Committee on Reactor Safeguards (see ML090710128), as well as by an external panel of independent experts (see Appendix B of NUREG-1806) and the Commission.

- 7. The NRC has classified Palisades as "one of the most embrittled" plants. If Palisades follows the NRC regulations is the probability of fracture still extremely low? If 10 CFR 50.61a is used by Palisades would the NRC consider Palisades "safe to operate" or would the NRC shut it down?**

Yes, as stated during the webinar, as long as operating reactors remain compliant with 10 CFR 50.61 or, if elected and approved, 10 CFR 50.61a, the probability of developing a crack through the vessel wall as a result of PTS remains extremely low. If Palisades elects to use 10 CFR 50.61a, and if the staff approves the submittal justifying this election, then Palisades fulfills NRC regulations with regards to PTS, is safe to operate, and there would be no basis for a shutdown.

- 8. The critical weld heat at Palisades is from the same heat as materials at Robinson and Indian Point. How does the amount of brittleness compare? Is this different that the reference temperature?**

Palisades recently performed an evaluation of Charpy V-notch data (i.e., a test that measures the energy absorbed by a material during fracture) from all surveillance programs in which the limiting weld wire heat (W5214) for Palisades was exposed (ML110060694). As noted in the question, these surveillance specimens were exposed to radiation not only in the Palisades reactor, but also in HB Robinson (Unit 2) as well as Indian Point (Units 2 and 3). These surveillance data showed a scatter (uncertainty) well within the bounds anticipated by the NRC's prediction formula (see Regulatory Guide 1.99, "*Radiation Embrittlement of Reactor Vessel Materials*," Revision 2 (<http://pbadupws.nrc.gov/docs/ML0037/ML003740284.pdf>)). The embrittlement data from specimens exposed in the Palisades reactor were somewhat below the mean trend for this weld wire heat, indicating that, if anything, the brittleness from the Palisades exposure was somewhat less than in HB Robinson and Indian Point (see ML13093A191). However, it should be noted that because these differences are all within the expected scatter they are not regarded as being statistically significant.

Reference temperature is the metric that the NRC uses to quantitatively assess brittleness, so these terms may be regarded as synonymous. Steel having a high "reference temperature" also has a higher degree of brittleness than steel with a low reference temperature.

**9. Did the Palisades power uprate which the NRC so readily approved worsen the neutron flux on the reactor pressure walls? Did NRC even consider the embrittlement and pressurized thermal shock risks of approving the power uprate?**

The NRC explicitly considered embrittlement and pressurized thermal shock in evaluating Palisades' power uprate. In 2004 the NRC issued Palisades a license amendment authorizing a 1.4 percent power uprate (see ML040970622, "Palisades Plant - Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate"). This power uprate demonstrated to the NRC the licensee's instrumentation met the accuracy requirements to monitor reactor power to allow the uprate. As can be seen in the following excerpt, taken from the previously mentioned document, the NRC considered both embrittlement and pressurized thermal shock in its assessment of Palisades' power uprate request.

*With respect to PTS events, the NRC staff previously approved revised neutron fluence values and the PTS assessment for Palisades by letter dated November 14, 2000. The licensee's measurement uncertainty recapture (MUR) Power Uprate Analysis Report assesses the impact of the power uprate on the neutron fluence values for the reactor vessel (RV) materials as a function of the impact the increase in fluence values will have on the effective full power days for the unit. This assessment indicates that the fluence values used in latest PTS assessment bounds the slight increase to the fluence values assumed for the MUR power uprate. Therefore, the most up-to-date PTS evaluation for Palisades is still valid even for the uprated conditions for the plant. (page 32)*

As can be seen in the highlighted section, the slight increase in the radiation (fluence) exposure that would result from the power uprate had already been accounted for in the regulatory estimate of embrittlement.

**10. When did 10 CFR 50, Appendix H become a rule? Was this before or after Palisades was licensed? What impact does this have on the requirements for Palisades to use surveillance capsules? Did Palisades use capsule surveillance data for its reference temperature calculations? If no, what impact did this have on the results?**

The requirement to implement a surveillance program to monitor the effect of embrittlement on the steels from which the reactor pressure vessel beltline is constructed is made in 10 CFR 50 Appendix H, which was issued in 1973 (Federal Register, Vol. 38, No. 136, July 17, 1973). Palisades received its operating license on February 21, 1971. Nevertheless Palisades implemented a surveillance program consistent with then-standard industry practice (i.e., implementation of the requirements of ASTM E185, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels". ASTM E185 has since been required by 10 CFR 50, Appendix H. Thus, even though Palisades entered service two years before 10 CFR 50, Appendix H, was adopted, Palisades' surveillance program was designed in accordance with ASTM E185 and it is effectively compliant with 10 CFR 50, Appendix H. As detailed in (ML110060694, "Revised Pressurized Thermal Shock Evaluation for the Palisades Reactor Pressure Vessel"), Palisades has used surveillance data as part of its reference temperature calculations. The use of ASTM E185 is part of the plant's licensing basis.

**11. But how long has it been since the last capsule was removed? Since the last capsule was analyzed? What if the embrittlement has taken place at a much more accelerated rate than NRC's modeling would predict?**

The last two capsules removed from Palisades were capsule SA-240-1 (removed in 2000) and capsule W-100 (removed in 2003). Capsule SA-240-1 contained the limiting (i.e., most embrittlement sensitive) weld from Palisades. Neither these, nor any other, surveillance data from Palisades provides any indication to the NRC of embrittlement occurring at a "*much more accelerated rate*," than is expected in Palisades, (see ML13093A191).

**12. If capsules were removed in the mid-1990s and 2000s, as NRC just said, that's a decade or two ago. Has the NRC simply extrapolated to predict the severity of embrittlement? What if NRC's understanding is flawed? What if the extrapolation is non-conservative? How can NRC speak with any confidence, if the last data collected – and very few data points at that – are over a decade old? This is not science. This is guesswork. The safety risks are too high for this lack of science.**

Following the requirements of ASTM E185 the surveillance capsules are designed to accumulate irradiation damage at a rate faster than that experienced by the wall of the reactor pressure vessel, which lies further away from the active core than do the surveillance capsules. Consequently the surveillance program provides measurements of embrittlement, or reference temperature, corresponding to a number of years of operation that is well in advance of the actual years of operation of the reactor pressure vessel. The practice is adopted with the specific aim of ensuring that regulatory decisions are based on embrittlement data that is interpolated, not extrapolated. The surveillance information that has been submitted to the NRC for Palisades is summarized in (ML110060694). This document demonstrates that data for the limiting weld wire heat (W5214) is available for a level of radiation exposure that exceeds by a factor of two that which is expected to occur in the Palisades vessel on the date in 2031 at which its extended license will expire. In summary, the important factor is not the date of surveillance capsule removal, but the total radiation exposure to which the specimens in these capsules are subjected. For Palisades, some data have twice the radiation exposure that the plant will experience in its extended lifetime (see ML13093A191).

The surveillance data allows the NRC to make licensing decisions with regards to PTS that are based on interpolations within the available data, not extrapolations beyond the data. Moreover, the greater body of evidence that is available from other operating reactors having steels of similar copper and nickel contents indicates that the limiting weld in Palisades is embrittling in a manner that is fully consistent with both physical expectations and empirical evidence. To provide further assurance that the NRC's predictive formulae are appropriate, the NRC staff participates in relevant codes and standards bodies (ASTM), as well as in national and international scientific conferences on these topics. Based on these experiences it is possible to state that there is no evidence available suggesting that the predictions of embrittlement trends for Palisades are incorrect. Finally, as a practical measure, NRC regulations require that Palisades, and indeed all operating plants, use an intentional over-estimate of the expected embrittlement trends when calculations are made to support the plants' licensing bases.

**13. Could you please explain how PRA is relied as a frequency of occurrence instead of as a predictor of occurrence? Currently, the NRC relies on PRA as a predictor – PRA is not.**

In a sense the terms "predictor" and "frequency" as described in the question refer to the same concept. PRA includes a "predictor" (frequency) of an occurrence of some event. The frequency term is a numerical quantity that represents how likely it is that an event will occur in the future. The frequency question is one part of the PRA process. More specifically, in the context of evaluating risk, PRA is commonly expressed as a "risk triplet" in that it attempts to answer these three questions:

1. What can go wrong (accident scenario)?
2. How likely is it (frequency on a per reactor year basis)?
3. What are the consequences (impact on the plant or on people)?

The NRC uses PRA models to look at the frequency and the consequences of NOT achieving safe shutdown conditions.

**14. Please provide duration under which 200 degrees sudden cool down criteria. Need a time frame.**

The statement was made during the webinar that a sudden cooldown, from operating temperature, in excess of 200 °F is needed to generate any non-zero risk of through wall cracking. This statement was based on an examination of all cooldown transients modeled in NUREG-1806 (<http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1806/>). This study addressed a wide spectrum of potential PTS events, including large diameter pipe breaks on both the primary and secondary side. For these large diameter breaks a 200 °F temperature drop would occur within 2 to 4 minutes.

**15. How much does it reverse the effects of embrittlement – annealing?**

The magnitude by which annealing reverses the effect of embrittlement is referred to as "recovery." Recovery depends on the annealing temperature and time, with temperature being the dominant factor. Two different procedures can be used to anneal a RPV, a wet anneal or a dry anneal. A wet anneal is performed with cooling water remaining in the RPV and it cannot be performed above the RPV design temperature of 650 °F. Annealing near this temperature results in low recovery; a reduction of the radiation-induced reference temperature of 10-30 percent is typical. A dry anneal requires removal of the cooling water and internal components along with application of heat to the inside of the vessel; it would be performed at temperatures in the range of 800-930 °F. A dry anneal would result in a recovery of approximately 80 percent of the radiation-induced reference temperature shift. Data shows that re-embrittlement after annealing occurs at a slower rate than occurred prior to annealing. However, the NRC regulatory guide on annealing (Regulatory Guide 1.162, *Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels*) conservatively assumes that embrittlement occurs at the same rate after annealing as it did before annealing.

**16. Are you going to be able to make pertinent measurements at those reactor vessels that are not in service anymore, such as Crystal River Unit 3 for example?**

The NRC has no regulatory requirements for embrittlement measurements on vessels that are no longer in service. In February 2013 the Duke Energy Company announced that it will not return Crystal River Unit 3 to service. No decision has been made to test samples removed from decommissioned reactors for embrittlement. However, the nuclear industry in Europe has pursued projects where small samples of ex-service vessels (e.g., Gundremmingen-A in Germany, Chooz-A in France, and Griefswald in the former East Germany) were removed and tested to measure embrittlement so that these measurements can be compared to the results of predictive formulae. These comparisons typically show that the predictions are accurate to within the scatter associated with the experimental measurements. While these experiments have all been conducted on European reactors, the embrittlement comparisons are appropriate to reactors in the USA. Also, the NRC participated in the Gundremmingen study, see NUREG/CR-5201, "Experimental Assessments of Gundremmingen RPV Archive Material for Fluence Rate Effects Studies," ADAMS ML111310052.

**17. How are the neutron flux predictions codes benchmarked for accuracy?**

Neutron fluence codes are qualified in accordance with Regulatory Guide 1.190, "*Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.*" Benchmarking is a three step process. The codes are benchmarked against operating reactor measurements using in-vessel surveillance capsule dosimetry, measurements made external to the vessel, or both. Although specific benchmarking to a plant of interest is preferred, it is acceptable to the NRC to use benchmarking from a plant of similar design. Next, the codes are usually benchmarked against a pressure vessel simulator benchmark (scale mock-up experiment), such as the Pool Critical Assembly at the Oak Ridge National Laboratory. Finally, the codes are usually qualified using a fluence calculation benchmark problem. An example of a benchmark problem appears in NUREG/CR-6115, "*PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions.*" This three-pronged approach ensures that uncertainties and errors associated with nuclear data, numerical methods, and plant-specific considerations, such as specific geometric representation of the reactor and vessel, are all thoroughly investigated.

**18. Can you discuss the primary difference between fracture toughness limits in 50.61 versus alternate requirements in 50.61a?**

As was stated during the webinar, 10 CFR 50.61a was developed as one option by which licensees could choose to show vessel integrity and safety should the level of embrittlement be projected to exceed that required by 10 CFR 50.61 within the plant's licensed lifetime. The limits on embrittlement, reference temperature, and fracture toughness (all of these terms may be regarded as synonyms) in 10 CFR 50.61a are less restrictive than those in 10 CFR 50.61; this being justified by greater accuracy in the models on which the 10 CFR 50.61a limits are based, by much greater and improved knowledge of both plant embrittlement data and plant operating procedures, and by benchmarking and validation of the models relative to scale experiments. The major factors that differentiate the fracture toughness model in 10 CFR 50.61a and 10 CFR 50.61 are summarized in the table below; the factors listed at the top of the table have the greatest quantitative effect of the difference between the reference temperature limits in 10 CFR 50.61 and 10 CFR 50.61a.

Component of Model	10 CFR 50.61	10 CFR 50.61a
Unirradiated reference temperature	RT <sub>NDT</sub> : an intentionally conservative representation	RT <sub>NDT</sub> plus correction, which is based on data, to account for the conservative difference between RT <sub>NDT</sub> and directly measured fracture toughness data
Flaws	Flaws intentionally larger than found in service, and all assumed to be found on the inner-diameter surface of the reactor vessel.	<ul style="list-style-type: none"> <li>• Flaws sized to be more representative of those found in service, based on both destructive and non-destructive evidence.</li> <li>• Flaws assumed to mostly be embedded in the reactor vessel wall, again based on evidence from service.</li> <li>• Number of flaws significantly over-estimated (a conservatism) relative to what is found in service.</li> </ul>
Fluence	All materials in the reactor vessel beltline assumed to experience the peak fluence that occurs anywhere on the inner diameter of the vessel.	The fluence associated with beltline materials is the actual fluence to which they are subjected. Fluence variation in the beltline is significant; it depends (primarily) on the water gap between the core and the inner diameter.
Effect of radiation on reference temperature	Uses equation from Reg. Guide 1.99 Rev. 2	Uses equation from 10 CFR 50.61a. This equation is based on over four times more embrittlement data from operating plants than Reg. Guide 1.99 Revision 2.

Taken together, these factors justify the less restrictive embrittlement limits in 10 CFR 50.61a **provided that** the licensee demonstrates that their data is consistent with the underlying principles of the NRC model used to develop 10 CFR 50.61a. These assumptions include (a) that small defects in the reactor vessel in question are accurately represented by the NRC's technical basis calculations, and (b) that the licensee demonstrates that the embrittlement trends in the reactor vessel in question are also accurately represented by the NRC's technical basis calculations.

Further details concerning the technical basis for the 10 CFR 50.61a embrittlement limits can be found on the NRC's website in NUREG-1806 (*Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)*) and NUREG-1874 (*Recommended Screening Limits for Pressurized Thermal Shock (PTS)*).

- 19. Have there been any preventative hardware installations to limit neutron exposure in the PWR fleet? If so, can you describe those projects and the extent of their success? And if not, what options are available for preventative hardware installs?**  
Preventative measures against PTS include the following:

- a) Many plants employ a technique referred to as “inside out fuel loading,” which is designed to reduce the neutron flux to which the reactor vessel is exposed and, thus, its degree of embrittlement. Following this strategy, new fuel assemblies are first placed in the middle of the core with low power assemblies on the core periphery. As the assemblies are used (or “burned”) they are moved, in subsequent refueling outages, toward the outside of the core and, thus, closer to the inner diameter of the reactor vessel. The partially spent fuel provides shielding to the vessel wall. This technique is used by Palisades.
  
- b) Neutron shield assemblies made of stainless steel are placed on the outside of the core. These also provide shielding for the reactor vessel, lowering the number of neutrons that escape the core and, thus, are available to embrittle the vessel steel. This technique is used by many plants, including Palisades.

In addition to answering questions from members of the public on March 19, NRC representatives agreed to provide an answer to technical questions regarding the topic of PTS that were submitted during the meeting, but were not answered during the allocated meeting time. The answers to these questions and the follow-up to two questions answered during the webinar are included in this meeting summary (Enclosure 2). Availability of a recording of the webinar will be addressed by separate correspondence.

Sincerely,

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John B. Giessner, Chief  
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