Comment to the Diablo Canyon Independent Safety Committee Regarding Discrepancies in Reported Fracture Toughness Test Results for DCPP Unit 1

A Joint Public Comment By

Bruce Severance, Regulatory Compliance Engineer
Member of the Diablo Canyon Decommissioning Engagement Panel (Commenting individually)

and

Dr. Digby MacDonald, Hon-FRSC, Hon-FRSNZ, FNACE, FECS, FISE, FIC, FASM, FWIC, Khwarizmi Laureate in Fundamental Science, Docteur Honoris Causa-INSA Lyon, Faraday Memorial Trust Gold Medalist, Gibbs Award Recipient (IAPWS), De Tao Master, Frumkin Medalist, 2013, OLIN Palladium Medalist, 2015, Ad Augusta Awardee, Auckland Grammar School, 2016, Professor in Residence, Departments of Nuclear Engineering and Materials Science and Engineering, University of California at Berkeley, Distinguished Professor Emeritus of Materials Science and Engineering, Penn State University

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Executive Summary

The following comments are offered with a great deal of humility and respect for the DCISC's accomplished expertise in this field. The 2003 material stress tests were required by ASTM E185-70 and Appendices G&H of 10 CFR 50, to monitor the continual loss of "fracture toughness"*[1a] of the Diablo Canyon Power Plant (DCPP) nuclear reactors. It appears that the stress tests (also "coupon analyses" and "capsule tech reports"), conducted on numerous material samples that represents the various components and welds within the reactor, reveal differing degrees of radiological embrittlement. Although there is some art in getting the many material sample failure test data from a given capsule to fit calculated predictions based on the degree of radiation exposure (via fluence calculations and dosimetry readings), there are in fact calculation cross checks that reduce the margins of uncertainty. An excerpt from page 6-5 of the Capsule V Report*[2b] (below), suggests the main way to assure accuracy is confirm a high correlation between predicted and measured values to keep the range of uncertainty to a minimum. Of course, this strategy presumes that either the measured data or the calculated data are correct. If both are incorrect, then the strategy is meaningless. Capsule Y Report (1993)*[1] has a higher degree of predicted-to-measured correlation than the more recent Capsule V Report (2003). In some cases, the Capsule V Report*[2a] predicted failure points that indicated a higher degree of embrittlement than what was measured, and one might argue in favor of revision to the fluence calculation to adjust for this

6.3 NEUTRON DOSIMETRY

The validity of the calculated neutron exposures previously reported in Section 6.2 is demonstrated by a direct comparison against the measured sensor reaction rates and via a least squares evaluation performed for each of the capsule dosimetry sets. However, since the neutron dosimetry measurement data merely serves to validate the calculated results, only the direct comparison of measured-to-calculated results for the most recent surveillance capsule removed from service is provided in this section of the report....

The measured-to-calculated (M/C) reaction rate ratios for the Capsule V threshold reactions range from 0.78 to 0.97, and the average M/C ratio is 0.89 ± 9.8% (1σ). This direct comparison falls well within the ± 20% criterion specified in Regulatory Guide 1.190; furthermore, it is consistent with the full set of comparisons given in Appendix E for all measured dosimetry removed to date from the Diablo Canyon Unit 1 reactor. As a result, these comparisons validate the current analytical results described in Section 6.2 which are deemed applicable for Diablo Canyon Unit 1.

Cap V measured to calculated data ratios within range specified by RG1.190 (20%)

“conservatism” by reducing the mathematically predicted radiation damage to better fit the measured data.
It is not immediately obvious to the average reader how a “fluence calculation” works or how they can precisely predict or model the total radiation exposure or “neutron fluence” of each material sample within a margin of uncertainty or even how the fluence is related to radiation embrittlement. The models are complex and three dimensional, (fast-moving neutrons flying in all directions) so some welds near the mid-section or “beltline” of the reactor will generally see higher rates of neutron bombardment than other components. However, the models are imperfect and because they are imperfect, they should be calibrated against experiment to obtain numerical values for poorly known model parameters. This is best done by constrained mathematical optimization. Over the years, hand calculations have been updated with more complex computer models with “libraries” of components and variables to simulate various operating conditions in three dimensions.

In reality, the material stress tests are also imperfect, offering results only within a range of uncertainty and deviation. However they offer modest repeatability. The more foundational fluence calculations that predict the level of radiation damage and embrittlement of all components throughout the reactor are used to predict the embrittlement of each component as it may be tested under standardized lab conditions. As suggested by NRC Regulatory Guides RG1.99 and 1.190, a high correlation between predicted values and stress test results is a reasonable means of confirming the accuracy of the fluence calculations, except for the caveat noted above when both sets of data are incorrect. Where it is possible that the data points for the stress tests may fall on a tight curve, stress tests can also have greater “deviation” and offer a “buckshot” of two-dimensional information to affirm fluence calculations. Nevertheless, RG1.99 defines five credibility criterion that define the acceptable parameters under such circumstances.

The PTL curve and the associated reports (PTLRs) are a more three dimensional guideline for the maximum pressures and lowest temperatures the embrittled reactor can sustain. The PTLRs are what actually define the safe operating limits of the reactor, with the highest risk of catastrophic failure occurring at low-temperatures but in scenarios wherein pressures are still high or may climb quickly. Such conditions are described as pressurized thermal shock (PTS). This condition can result from any major pipe break in the primary coolant system or stuck valve that results rapid loss-of-coolant accident. A rapid loss of coolant automatically triggers the “Emergency Core Coolant System” (ECCS) to introduced sometimes large amounts of cold water, usually not preheated and kept at ambient outside temperatures (50°F to 80°F). The “ductility” or elasticity of the 8” thick steel walls of the reactor vessel and the many welds over time become more embrittled due to displacement of iron atoms, a process that can be mathematically modeled. Gradual embrittlement leads to the formation of hairline cracks which can be harmless at normal operating pressures and temperatures. These can be monitored using ultrasonic testing (UT) and other means of in-service inspections (ISI), and given the potential failure modes, such inspections should occur frequently in older plants with questionable embrittlement. We will try to make these concepts more comprehensible in the following text, some in less technical terms for general audiences, and some sections more technical. In our opinion, RG1.99 falls short in not accounting for the approximately half-inch thick, ductile austenitic stainless steel liner and the possibility of hydrogen injection into the RPV from the radiolysis of the coolant that contains considerable hydrogen [typically 25 cc(STP)H_2/kg H_2O]. In the case of the Davis Besse PWR a few years ago, the RPV had been breached at the control rod guide tube penetrations via corrosion by concentrated boric acid, yet the reactor continued to operate at full power with the coolant only being contained by the stainless steel (SS) liner. In the case of atomic hydrogen injection, hydrogen embrittlement is a well-known phenomenon in many other technological areas including the oil and gas industry (embrittled heat-affected zones welds in production tubing), naval aviation (embrittled landing gear), and bridges (e.g., failed high-strength steel tendons in the new Bay bridge a few years ago), to name but a few. In our opinion, the former (SS liner) will likely mitigate the RPV radiation embrittlement phenomenon but the latter will certainly exacerbate the problem. It is for this reason that we describe the latter as a "force multiplier".

Given these cross-checks, one would expect "tighter" uncertainties and margins of error on the flux calculations, but uncertainties remain and must be accounted for mathematically. The 1993 Capsule Y Report suggests combined uncertainty in the range of 13%-15%. However, one may encounter larger ranges of uncertainty, or larger deviations in predicted values wherein there is some art to achieving a "best fit curve". There are those scientists that acknowledge there is as much art as science in achieving a high predicted-to-measured (calculated versus stress test) correlation, but this remains the best means of producing the most credible PTL curves and reports which define hard limits to the pressures and temperatures that reactor vessel can sustain. With something as important as projecting the total radiation damage and therefore the embrittlement of reactor materials, one would not expect margins of error for a fluence calculation to exceed 20% to 30%. In short, the ability to correlate differing calculation methods increases accuracy and a fluence calculations (estimate of radiation damage) can and should be "adjusted" to allow the predictions stemming from them to more accurately fit measured stress test values provided that the adjustments are justified physically. Such a methodology, as defined by the NRC Reg. Guides, would increase the predicted-to-measured correlation, and would possibly reduce uncertainties.

The greater the uncertainty of the radiation damage predictions, the greater the "margins of error" or "safety margin" that a regulator or reactor operator would want to account for by reducing projected life of the reactor. In theory, if there is a greater uncertainty of the fracture toughness of the reactor vessel, this uncertainty should be reflected in a corresponding reduction in the predicted life of the reactor, thereby reducing its projected "effective full-power year" rating (EFPY). In some cases, there may be too few data points for a thorough probabilistic risk assessment, which in turn raises questions as to how far on the side of caution the NRC and the operator wish to err.

Despite PG&E's and Westinghouse's level of stated confidence in the 2003 Capsule V Report and associated fluence calculations, PG&E and Westinghouse currently argue that the data in all three capsule reports to date (S, Y and V) should be deemed "not credible" based on non-conformance with Reg. Guide 1.99, Rev 2, Criterion 3 which states:

> ***"Where there are two or more sets of surveillance data from one reactor, the scatter of the ΔRTNDT values about the best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail to meet this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy [USE] if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.***

It appears from the excerpt below that PG&E misinterpreted the above criteria to restrict the "scatter" or deviation to one standard deviation of 28°F for welds and 17°F for base metal. Based on this interpretation, they use Criterion 3 to deem the 2003 stress test data are "not credible'. However, PG&E seems to note correctly the 2-sigma deviation allowed in their 1993 Capsule Y report, even though they argue for over a decade in their correspondence to the NRC (2003 to 2016) that the Capsule V data did not meet the RG1.99 Criterion 3. The excerpt below from the 1993 Capsule Y Report*[3] states that PG&E interpreted RG 1.99 as defining deviation for welds as 56°F rather than 28°F to be within credibility limits:

Capsule Y Report, Pg. 1-2

- **The surveillance Capsule Y test results indicate that the surveillance weld metal 30 ft-lb transition temperature shift is 10°F greater than the Regulatory Guide 1.99, Revision 2 prediction. This increase is bounded by the 2 sigma allowance for shift prediction of 56°F. The EPRI Hyperbolic Tangent Curve Fitting Routine measured average upper shelf energy decrease for the weld metal is 1 ft-lb less than the Regulatory Guide 1.99, Revision 2 prediction and the ASTM E185-82 measured average upper shelf energy decrease is 5 ft-lb greater than the Regulatory Guide 1.99, Revision 2 prediction.**

Capsule Y Report, Pg. 1-3

- **Based on the criteria given in Regulatory Guide 1.99, Revision 2, the Diablo Canyon Unit 1 Surveillance Program is judged to be credible.**

Excerpts from Pages 1-2 & 1-3 of the 1993 Capsule Y Report

PG&E makes the argument from 2003 into 2016 that all their surveillance data are "not credible" based on a 1-sigma scatter rather than 2-sigma. This may seem trivial, but this 1-sigma versus 2-sigma argument is the key to PG&E's ability to throw out their stress test data, and decouple their fracture toughness projections from any attempt to correlate fluence calculations and their predictions to the lab data. It allows them to revert back to a pure calculation or substitute data from a "sister" power plant, provided operating conditions and metallurgy are identical. Despite the NRC raising concerns about their interpretation and their recitation to PG&E of the criteria, PG&E never fully addresses the obvious misinterpretation. All of the metal sample stress test data actually fall within this credibility range as well as meet all other credibility criteria including RG1.190, as stated in the excerpt from the Capsule V report on page 1, above (Neutron Dosimetry). In short, according to regulation, the surveillance data should not be entirely ignored.

This repeated misinterpretation of RG1.99 Criterion 3 is especially concerning because some of the stress test samples exhibited "fast fracture" (indicating brittleness). Under NRC RG1.99 Position 2.2, the projected fracture toughness of the reactor through end of life is based upon physical evidence, "credible test data from two or more capsules". If you are able to deem this evidence as "not credible" you are allowed to revert to RG1.99, Position 1.2, which allows an alternative calculation method to determine the brittleness of the reactor vessel materials without the use of stress test data.

PG&E states in their cover letter to the 2003 Capsule V Technical Report (DCL-03-052) that:

> ***"<u>For the Unit 1 end of operating license (EOL) at approximately 32 effective full-power years (EFPY) on September 22, 2021,</u> the limiting RT_{PTS}*[4a] values calculated and their respective 10 CFR 50.61 screening limits are:***
>
> ***RT_{PTS}(weld 3-442C) = 250.9°F, which is <270°F plate or axial weld limit***
> ***RT_{PTS}(weld 9-442) = 192.8°F, which is<300°F circumferential weld limit***

Therefore, the PTS screening limits are met at EOL [40-year end-of-life]. PG&E performed this evaluation." *[4b]

At least at the time that this letter was sent to the NRC along with the 2003 Capsule V Report (WCAP-15958-NP), PG&E had reached the conclusion that Unit 1 was going to reach fracture toughness limits with enough certainty to publish this statement to the NRC. Their report seems to suggest that they are projecting a date in September 2021 as a reasonable estimate as to when the Unit 1 reactor will approach the fracture toughness limits. The Capsule V Report data revealed that a "limiting" (most compromised) weld number 3-442C had a failure temperature of 250.9 °F, just under the allowed threshold of DBTT<270 °F*[4C] (to be defined below and in footnotes), at only 32 EFPY (effective full-power years). This projection was extrapolated to 43 EFPY in the 2009 License Renewal Application (LRA), perhaps a reasonable extrapolation that would have suggested a generous margin of compliance at the projected 40-year life (43EFPY), but not for the 60-year life (54EFPY) which they later asserted). We will try to break down the implicit mathematical uncertainties inherent in the method by which the Unit 1 reactor vessel projected life was revised from appearing to reach limits in 2021 to an interpretation of the data in the 2011 LRA Update that indicates Unit 1 is fully compliant for the full 20-year extended period. The stages of the logic applied, and the regulatory compliance paths that were employed are worthy of investigation.

At the time the License Renewal Application was submitted to the NRC in 2009 (2009 LRA), PG&E speculated that they would be able to meet compliance requirements with Unit 1 under the 10 CFR 50.61a (50.61 alternative calculation method) for which new rules were then in development and about to be approved by the NRC. In short, 50.61a is less onerous than complying by means of 50.61, but it required that reactor vessel materials meet ASTM material specifications, and Unit 1 was known to have metallurgical flaws, (high copper and nickel impurities in weld material Heat No. 27204). For whatever reason, it appears that PG&E was not able to meet the required criteria of 50.61a because by early 2011, they had fallen back on the more onerous RG1.99 Position 1.2 under 10 CFR 50.61. Under this method, PG&E is allowed a complete revision of the fluence calculation (document numbers WCAP-17299-NP & WCAP-17315-NP, which PG&E deemed as confidential) and then on this basis, PG&E appears to revise the stress test data in their 2011 Update to the License Renewal Application (DCL-11-136, pages 80-86) to project DBTT and Upper Shelf Energy values (USE=a measure of fracture toughness) that meet the minimum fracture toughness requirements for both Unit 1 and 2 at the end of a 20-year extended operation. One must ask how we get from barely complying with Unit 1 through 2021 to meeting all requirements through 2044?

Reg. Guide 1.99 actually states the following in regard to stress test data credibility:

> **"When surveillance data from the reactor in question become available, the weight given to them relative to the information in this guide will depend on the credibility of the surveillance data as judged by the …criteria."** *[5]

This would seem to suggest that if the stress test data is credible, it should be referenced, and there is some PG&E correspondence that suggests they interpreted RG1.99 to mean just that. Should the DCISC choose to investigate the data in its detail, they would find that the data in all the Capsule Y and V reports are within the definitions of the five credibility criteria defined in RG1.99. There is no cause to deem any of the stress test data "not credible". It is questionable that PG&E working with Westinghouse has persisted in doing so even after the NRC raised concerns.

It should be noted that there is no correspondence on the NRC docket that confirms NRC approval of the proposed approach without qualifications or requests for further clarification. Certain paragraphs may appear to suggest NRC approval of the approach, but the NRC never received a copy of the "confidential" fluence report upon which the calculations are based. To the contrary, there are a number of Requests for Additional Information (RAIs) that raise concerns, and a 2015 letter in which PG&E states there were some nozzle shell welds that did not meet fracture toughness limits. These issues are not resolved in the exchange leading up to the withdrawal of the LRA in June of 2016, to be discussed in more detail below.

The text of the 2011 License Renewal Application Update (DCL-11-136) indicates that PG&E used their new radiation damage calculations (WCAP-17299-NP) to extrapolate new failure temperatures. Based on this new, confidential methodology, PG&E justifies revision of the physical stress test data for the most limiting weld and plates. A critical weld #3-442C failure temperature of 280 °F is revised to only 243 °F, from above the ductile-brittleness transition temperature limit (DBTT of <270 °F) to below this critical threshold, (See Appendices B & N*[6]).

Notably, only the data from the most limiting (most embrittled) components in these tables shifts by 15% or more, while other values move in the range of only 1%. appearing to adjust only those components most at risk of failure and non-compliance. The fluence calculations have shifted by more than two orders of magnitude (from 1.37×10^{19} n/cm^2 to 1×10^{17} n/cm^2). Although one may buy a reasonable argument for adjusting for "conservatism" in the older fluence calculation methodology, it is important to remember the high predicted-to-measured correlation of both the Capsule Y and V estimates of radiation damage. Given this high correlation, it is counter-intuitive that the fluence calculations would be off by orders of magnitude. When predicted values correlate with the measured lab-test data, the margins of uncertainty cannot be off by orders of magnitude. In regard to the 1993 and 2003 stress tests, uncertainty was in fact already within the credibility range dictated by RG 1.190 as noted above in the 2003 Capsule V Report ("Neutron Dosimetry excerpt on page 1).

One would expect some range of "uncertainty" in all of the equation variables contained in the capsule reports to be fairly modest, and not exceeding a combined total of 20% to 30%, including fluence factors, formulas, CF factors (copper & nickel impurities) and the possible random fracture of samples based on localized impurities. The raw data as well as calculation methods concealed in confidential reports (WCAP-17299 & 17315) were only revealed by a FOIA filing on July 5th and would call for third party review of the mathematical models to confirm the ranges of uncertainty. Due to the claim of confidentiality, sufficient data has not been available up until this time.

One might expect to find one test sample out of many to be farther out of range on occasion, given the details of how the Charpy V-Notch stress test equipment works. The machining of the samples, the size or mount variation of the samples, the consistency of the swing of the pendulum hammer, etc. as well as irregularities in the distribution of impurities may cause one sample to appear more embrittled than it may actually be. However, the test is standardized precisely to offer repeatability, and such deviations in the data are not usual. For example, even if actual chemical testing produces slightly different results than tabular information on chemistry factors, such as the standardized CF tables contained in RG1.99, one would assume such reports were conducted with a thorough attention to detail, and they would not be off by orders of magnitude. The acid test of the accuracy of the predictions is how closely they correlate with the measured stress test results and if this correlation falls within a narrow range in accordance with both RG1.99 and RG1.190 the original capsule tests should be deemed credible. There is some likelihood given the calculations that are visible in the public NRC docket, that PG&E's final compliance strategy based on confidential reports, suggests a methodology with much higher degrees of uncertainty. Without the thorough review of the confidential reports this can only be extrapolated. However, there are concerns about the methodology even prior to such a review.

For example, one would not expect that a material that fails at 280 °F can be "recalculated" to fail at a different temperature. The Regulatory Guides suggest adjusting the fluence calculations to better fit the data (predicted-to-measured correlation), but they do not suggest the opposite, adjusting the data tables according to new predictive models. ***There is in fact an inherent epistemological error in using a calculation method to alter the physical material stress test data. The problem arises when the methodology is not explained and changes to the data tables are made without fully disclosing the justification including footnoting every future reference.*** One doesn't need to be an expert in fluence calculations, to question the general methodology. How can one get from barely complying with fracture toughness requirements at 32EFPY and a projected shutdown date of September 2021, to complying with wide margins at 54EFPY (60-year operation)? It's a reasonable question. Considering the context of all three capsule stress tests, the fracture toughness of the reactor vessel seems to be on a trend line which then suddenly flattens or decreases. There are no known physical or mechanical principles that would cause radiation damage to cease despite continued radiation exposure.

The answer to ***How*** PG&E was able to radically shift the fracture toughness estimates lies in the so-called "confidential reports" that they have not submitted to the NRC (WCAP-17299-NP & WCAP-17315-NP) and which may not have been given to the DCISC with the benefit of footnotes clarifying revisions to stress test data. Any regulatory body should know how math was used to alter data and how such changes were justified. There is cause to be concerned about the methodology and details on this issue as there is physical evidence that is credible, and would suggest that Unit 1 is already past its fracture toughness limits.

Despite PG&E's 68% increase in the projected life of Unit 1 in addition to a 15% shift in the brittleness estimates of the most compromised plates and welds, there were still concerns expressed by PG&E as late as October 2015 regarding the nozzle shell welds. A statement on page 36 of DCL-12-124 admits that the nozzle shell welds and related components may not meet fracture toughness limits through the entire 20-year extension- even after the fluence calculations were used to justify an approximately 80% shift in the data:

> ***"For license renewal, Westinghouse performed additional calculations to define which materials… other than beltine materials, are projected to exceed the threshold neutron fluence of 1 x 10^{17} n/cm^2 at 54EFPY… The result of these …are documented in WCAP-17299-NP for Units 1 & 2, through ELOE [end-of-license-extension]. For both units, although the nozzle shell course and the associated nozzle shell to intermediate shell welds are projected to exceed 1 x 10^{17} n/cm^2 threshold, the nozzles themselves as well as the nozzle to nozzle shell welds remain below the 1x10^{17} n/cm^2 through 54EFPY."***
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> (DCL-12-124*[7], See p.36 & 50 for contradiction, See Appendix F for complete page 36 excerpt)

This PG&E notation that the nozzle assemblies have a compliance problem is contradicted in the same report where on page 50 PG&E states:

> *Pressurized Thermal Shock*
>
> All of the beltline and extended beltline materials in the Diablo Canyon Units 1 and 2 reactor vessels are projected to remain below the PTS screening criteria values of 270°F, for axially oriented welds and plates / forgings, and 300°F, for circumferentially oriented welds (per 10 CFR 50.61), through EOL (32 EFPY) and EOLE (54 EFPY).

The inconsistencies within the same report seem to further invalidate use of the "adjusted conservatism" argument. The reactor is either within limits for the projected 20-year extension, or it is not. It can't be both. Such inconsistencies undermine the credibility of the new fluence calculation strategy and the conclusion of compliance that is often repeated by PG&E. In short, PG&E claims to meet the requirements of 50.61 and RG1.99 Position 1.2, but even after a 2-magnitude shift in the fluence which reduces their predicted-to-measured correlation, they still have a compliance problem with the nozzle shells on both reactors for the through the end of extended operations (54EFPY).

The regulations do not appear to imply that a new fluence calculation can be devised that will allow a shift in physical stress test data if that data is "credible". If a metal sample breaks at 50lbs. of force at 280 °F, slightly above the DBTT limit of <270 °F, there is really no calculation that would justify alteration of this physical fact. If data from similar reactor vessels is being substituted, that would normally not be allowed if the local reactor data is deemed credible, and even then, it would only be allowed if the metallurgy and operating conditions are demonstrably identical. If the mathematical predictions arising from the fluence calculations are too high or too low, it is more "acceptable" to adjust the fluence calculation in order to find the "best-fit curve" to match the physical stress test data –not the other way around.

Background

It is common knowledge that there are known metallurgical flaws in the Unit 1 reactor vessel, excessive copper and nickel impurities in welds and plate metals that were discovered only after the Unit 1 RPV was delivered to DCPP. It is well documented that there were engineering errors made in the metallurgical specifications of Unit 1 plate and weld alloys and that Westinghouse, the manufacturer, realized their errors and corrected them prior to the second reactor vessel being installed at DCPP. As stated in a Fairewinds and Associates report filed with the CPUC in 2016:

> ***"Diablo Canyon Unit 1 was one of the first US atomic reactors ever designed and manufactured by the nuclear power industry, therefore, unusual, and consequential errors were made in the design and engineering. The wrong material was used to weld the atomic reactor vessel introducing impurities in the weld material that have caused significant and accelerated radiation damage in the form of embrittlement…Diablo Canyon now ranks as one of the five worst reactors out of the 99 remaining operational reactors in the US."**** [8] It is also the only nuclear power plant in the nation that sits atop a web of active fault lines.

For the non-technical reader, a background explanation of terms is offered here. The degree of radiological embrittlement at 40 years or 60 years of operation might be mathematically predicted using a mathematical model and verified based upon stress tests on metal samples when the plant has been in operation for only ten to twenty years. This is, in part, done by exposing the samples to higher radiation levels than the reactor vessel walls to accelerate the radiation damage of the samples, and employing a "fluence calculations" which determines within a range of "uncertainty", the extent of radiation exposure and damage by predicted end-of-life dates (32EFPY, 54EFPY, etc.). The metal samples can be heated to different temperatures and hit with a repeatable hammerlike force in a device that makes these tests controlled and repeatable, a Charpy V-Notch stress test. The predicted-to-measured values can be correlated for accuracy. Other calculated cross checks are also used. However, no two pieces would break identically in the same test, so there is a degree of variation in the results often referred to as "deviation". The degree of deviation on a graph can look like a shotgun blast of data points and this is referred to as "scatter", but it generally follows a trend line which can be drawn through the scatter to find a "best-fit curve". The metal stress test samples can simulate or approximate the damage projected when the reactor vessel approaches end-of-life (EOL) or end-of-license extension (EOLE).

In 2003, PG&E performed the third and last required metal stress tests on metal samples (called coupons) that by design are left within "capsules" (square tubular enclosures) which are positioned in the reactor vessel in various locations in a manner that approximates radiation exposure for a range of "effective full-power years" (EFPY). The extent of radiation exposure or "neutron bombardment" that the vessel walls and welds experience can be measured using radiation sensors (dosimetry) that measure the instantaneous "neutron flux" as well as the total projected radiation exposure over the life of the reactor "neutron fluence". The neutron fluence is a projected estimate of total radiation damage to the steel alloys in the vessel, and can be calculated based on a range of projected lives (32EFPY is approximately 40 years if you add schedule outages, and 54EFPY is approximately 60 years). There is more than one method of calculation, so different calculations can be combined to increase the accuracy of the projections. Also, since the fluence formula is a foundational calculation that underlies predictions of material degradation, it can thus be used as the basis for predicting the results of material stress tests within a range of accuracy. It is acceptable in the regulatory protocols to modify the fluence calculation to adjust for inaccuracies in the dosimetry and modeling to better fit the stress test data and the "best fit curve" defined by it.

For the lay person who may not immediately see the connection between temperature, pressure, and fracture toughness or brittleness, a similar and more-well know example of such an embrittlement failure mode is what sank the Titanic. At the time the ship was built in Belfast, Ireland, there had been a shortage of adequately annealed steels, and the less elastic steels that were substituted became very brittle in the cold Atlantic water on that fateful evening in April 1912. The ship had been thought to be "unsinkable" because the hull was designed with numerous compartments that were intended to isolate the flooding in the event of a hull-breach. If the ship had struck the iceberg directly on the bow, the blow would have probably penetrated the first few compartments and the ship would have returned to port for repairs. However, a study conducted a few years ago revealed that the captain's attempt to turn the ship at high speed caused a glancing blow precisely at the waterline where the steel hull had become embrittled by the cold water. The steel fractured (shattered) in a manner that many compartments separated by bulkheads flooded causing the front third of the ship to become too heavy for the brittle structure to sustain. Think in terms of a popsicle stick with a five-pound weight on one end of it. The bow broke off with the weight of the water filling it – and the rest is history.

Similarly, reactor vessel steels which are designed to operate under very high temperatures and pressures (>600 °F and over 2000 psi) become much more brittle at lower to mid-temperatures due to radiation damage as well as hydrogen embrittlement although the role of the latter is seldom acknowledged. The more brittle irradiated steel loses its elasticity over time and it increases in its susceptibility to fail at higher temperatures through brittle fracture. Regulation requires that this loss in "ductility" (elasticity) is measured precisely over time (10CFR 50, ASTM E185-70 & 82), and the methods for testing are standardized in every detail to minimize uncontrolled variables. The stress tests include a standardized "Charpy V-Notch" device that will exert a blow at a selected force with the sample heated to a specific and controlled test temperature. The sample must be precisely machined to specific dimensions to meet the test requirements. The standard limit for most reactor components (except some circumferential welds) accounting for some margin of safety is a DBTT of <270 °F. For this reason the "ductile-brittleness transition temperature" (DBTT) of <270 °F is defined (see again footnotes 4a and 4c). Samples breaking over this temperature fail the Charpy stress test standard because they exhibit brittleness at higher temperatures when heated steels should behave in a more ductile manner than cooler samples.

However, very few samples break at 269 °F or 271 °F and the spread of data resulting from the tests and RG1.190 would suggest ±20 °F as a reasonable margin of error to account for real-world variations, variability of impurities in the alloy, etc. The fluence calculations on the whole can only change the estimated radiation damage of the sample (predicted value) based on a mathematical model, (only an estimate) of accelerated radiation damage. Since this estimation of radiation damage is as much art as science, different methods of calculation, dosimetry (instantaneous radiation levels), and high correlation between predicted and measured test values are all used to cross check the accuracy of the fluence calculations. The projected end-of-life stress test "behavior" of a material sample, can exhibit either ductile (elastic) characteristics or "fast fracture" (more brittle). A fluence calculation that departs from the range of credibility of calculated cross checks or that doesn't fit credible data, merits re-evaluation. Coming up with a new methodology to "model" or calculate the brittleness of the Titanic's hull cannot change the brittleness of the hull when the embrittled metal failed or raise the ship. Science doesn't work that way.

The Questionable Accuracy of the Secretive Fluence Calculations

Fluence calculations can be moderately affected by many factors, including changing the fuel loading configuration within the reactor which may lower the thermal efficiencies of the plant but can extend fracture toughness projections and operational life. It appears PG&E began to change the fuel configuration in about 2011 in order to allow a longer operating life for Unit 1, although the precise date does not appear to be confirmed in their correspondence to the NRC. (FSARs have not been fully investigated.) However, such a change in plant operating conditions may buy a few years, credibly revising 32EFPY to 43EFPY for example. However, it is not likely that it could by itself explain large changes in the expected life of the plant such as a 20-year extension. Similarly, chemical testing of the plate alloys and weld material as opposed to using tabular data, would generally make only minor shifts in the data. After PG&E and Westinghouse completed the new fluence calculations in secretive Westinghouse Reports (WCAP-17299-NP & WCAP-17315-NP), PG&E appears to have asserted that they were so compliant with fracture toughness requirements that they no longer needed to change the fuel configuration (compromises thermal efficiencies and output), and further, their reports suggested that they no longer required RPV Ultrasonic Testing (UT) on the reactor vessel through the 60-year extended operation period (EOLE). UT would have been required under 10 CFR 50.61a the less stringent compliance methodology, and one must ask, why not perform extensive ultrasonic inspections of the reactor vessels and track the growth of hairline cracks over time? The 2011 Annual Update position on UT may have since changed, and this remains to be confirmed.

From the revised stress test data tables in the 2011 Update (DCL11-136 pages 81-85, Appendix N), it seems clear that the fluence values and embrittlement predictions do not change consistently from sample to sample. The result of the WCAP-17299 fluence calculations are counterintuitive, and appear to be out of range with credibility tests and margins of error (±20 % margins of error defined by RG1.190).Notably, the DBTT numbers move from barely complying at 32EFPY to fully complying at 54EFPY a 68 % increase in effective full-power years. The large shift in the revised 2003 test sample data from DBTT from 280.4 °F to 243 °F, concurrent with large changes in EFPY, merits explanation and further investigation.

There were, it appears open questions and requests for additional information (RAIs) that were left open and unanswered when PG&E elected to terminate the license renewal in 2016. The last correspondence from the NRC to PG&E dated February 2016*[9] requested the following:

RAI 4.2.1-2 ML16011A365 - NRC RAI2-2016

Background:

Attachment 2 of the applicants 2011 annual update (December 21, 2011) indicates that for both units, the nozzle shell course and the associated nozzle shell to intermediate shell weld are projected to exceed the 1×10^{17} n/cm^2 threshold. The applicant also stated, however, that the nozzles themselves as well as the nozzle to nozzle shell welds remain below the 1×10^{17} n/cm^2 threshold through 54 effective full-power year (EFPY).

Issue:

It is not clear to the staff why the nozzle shell course and the associated nozzle shell to intermediate shell weld are projected to exceed 1×10^{17} n/cm^2 while the nozzles themselves, the nozzle to nozzle shell welds, and the lower shell to lower head weld remain below the 1×10^{17} n/cm^2 threshold through 54 EFPY.

Request:

Identify the specific nozzles and nozzle-to-nozzle weld components that are being referred to in the above statement and identify what the inside surface neutron fluences are for these components, as projected to 54 EFPY. If any of the neutron fluences for these components are projected to exceed a value of 1×10^{17} n/cm^2 (E > 1.0 MeV) at 54 EFPY, provide the associated pressurized thermal shock (PTS) and upper shelf energy (USE) calculations for the components at 54 EFPY.

It is important to note that the above PG&E identification of weakness in the nozzle shell intermediate welds comes after PG&E revised fluence formula values downward by two orders of magnitude as well as lowering the fluence factor from 1.37 to 1.0, significantly below where the projections in the 1992 Capsule Y Report (1.05), (1.37×10^{19} n/cm^2 to 1.0×10^{17} n/cm^2). Despite the large adjustments in the fluence calculations, the nozzle shell structures did not meet fracture toughness requirements through the extended operation of the plant *for either Unit 1 or Unit 2.*

If the best-fit curve principle of the Reg Guide 1.99 procedure were interpreted accurately, the Capsule V Tech Report should have revised the fluence factor to approximately 1.15×10^{19} n/cm^2, in order to make the predicted and measured values on the Capsule V data more closely correlate on Table 5-10 of that report. There is no obvious justification for how the new 2011 fluence calculation was used to rewrite the Capsule V stress test data. Details are not explained. The employed methodology is omitted. .

An Important Revelation by the NRC Librarian

Third party access to the secretive WCAP-17299-NP fluence calculations has been of central concern in settling the questions of which credibility criteria is most applicable and whether PG&E's calculation method holds water. A conversation with a librarian at the NRC's docket office in mid-June 2023 revealed that the so-called "proprietary" document in question appears neither on the public-facing docket not on the NRC's private servers. The librarian, informed one of the coauthors that if it was not on the private server, it is not likely that the NRC has ever had possession of the document. One must ask how it is possible that the NRC never received the final update to the fluence calculations which would appear to be critical to the final approval of PG&E's fracture toughness assessment? Why would PG&E withhold submission of the full fluence report to the NRC? The record does reveal that the NRC issued a number of RAI's (requests for additional information) wherein they specifically asked for clarification of fluence calculations and the justification for using RG1.99 Criterion 3 to deem the surveillance test data "not credible" (See RAI 4.2.1-1 in Appendix K*[10]).

Remarkably, after PG&E asserted on numerous occasions that Westinghouse had deemed the WCAP-17299 report to be confidential and proprietary, the librarian found reference to the Westinghouse WCAP number and confirmed that the "NP" designation meant that it was, at least according to Westinghouse "not proprietary". Reference to the paper by title: WCAP-17299-NP, "Fast Neutron Fluence Update for the Diablo Canyon Unit 1 and 2 Pressure Vessels", Revision 0, February 2011 appears in only one place over the 5-year correspondence exchange between the NRC and PG&E. Given the fact that the new 2011 fluence calculations and the data table revisions in the 2011 LRA Update (DCL-11-136, pages 80-86) coincided closely with an important DCISC's publication, it is important to ask whether the revised data formed the basis of the DCISC's fracture toughness review articulated in their 2011 report entitled: "Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor"*[11] (Feb. 15, 2011). If the DCISC has only been shown the revised data and not the original 2003 Capsule V Report, it would explain how their report seems to reach an opposite conclusion that did not take into account all of the NRC requests for additional information at the time, which seem to inquire into a number of unresolved issues. The DCISC 2011 report concludes that based on their review of all the relevant reports to date, and pending NRC approval, both reactors look safe to operate through the 60-year extended operation period. At this point, without further approval by the NRC for PG&E's fracture toughness calculations, this DCISC report conclusion would seem premature.

Questionable Conclusions in the DCISC's 2011 "Evaluation of Pressurized Thermal Shock…"

The 2003 material stress test data and PG&E's cover letter indicated that DCPP Unit 1 may already be past its fracture toughness life expectancy by 2021.*[12] The 2009 LRA credibly revised the data to reflect 43EFPY (effective full-power years)*[13] but DCL-12-124 (2012) later stated that Westinghouse performed additional calculations which identified "materials [nozzle shell welds] in the DCPP pressurized vessels other than beltline materials, are projected to exceed the threshold neutron fluence of 1×10^{17} n/cm^2 at 54 EFPY (extended beltline materials)".*[14]

Despite PG&E knowledge of these identified issues, the DCISC states in their 2011 "Evaluation of Pressurized Thermal Shock…" that they "have reviewed DCPP reactor vessel surveillance data and analyses and other DCPP and NRC information… [regarding] application for a 20-year license renewal, [and have] … found no technical issues. Based on this we conclude that the two units (reactors) can both operate out to 60 years (if the NRC grants the two units permission to do so)."*[15] Although some NRC correspondence suggests that they may approve the application for the desired 20-year extended period (54EFPY, 60-year life), it does so with some qualifications and questions regarding justification and clarification on calculation parameters and values*[16] (Appendix H).

If the NRC has yet to approve the basis for the revision to the 2003 stress test data and has not yet reviewed the fluence calculations (in WCAP-17299-NP) as is required under 10 CFR 50 and ASTM E185-82, there is some possibility that Unit 1 will not be qualified for continued operation by the NRC. This becomes even more likely if the NRC recognizes the 2-sigma versus 1-sigma interpretation of scatter limits defined by RG1.99 Criterion 3. As long as there are unanswered questions about the extent of the nozzle shell weld embrittlement, the Unit 1 and Unit 2 reactors may currently be operating out of compliance with these regulations. There doesn't appear to be an unqualified approval by the NRC for PG&E's fluence calculations, fracture toughness estimates and pressurized thermal shock calculations for both units at Diablo Canyon Power Plant (DCPP).

The Credibility of the 2003 Capsule V Test Data

The fact that some samples representing weld 3-442C (Weld Heat No. 27204) were among the samples that exhibited "fast fracture"*[17] (significant embrittlement), is a sufficient cause for concern and the DCISC should investigate whether the surveillance data meets RG1.99 credibility Criterion 3 to determine whether the material stress test data are "credible" on this basis alone. PG&E's interpretation of this criterion, (presented above), states that the scatter or deviation should not exceed 28 °F for welds and 17 °F for plate materials, when the criteria actually states "double those values (56 °F for welds and 34 °F for plate materials). The data do not appear to exceed these limits. In any event, the DCISC investigation and interpretation of the RG1.99 credibility criteria and either validation or invalidation of the surveillance data is clearly warranted.

The following excerpts and tables from the Capsule V Report*[18] indicate samples that experienced a sudden drop in load, indicative of fast fracture, an indication of embrittlement. If some material samples exhibit fast-fracture characteristics, it is even more important that the material stress test data NOT be discredited without sufficient cause.

5 TESTING OF SPECIMENS FROM CAPSULE V

5.1 OVERVIEW

Testing was performed in accordance with 10CFR50, Appendices G and H[2], ASTM Specification E185-82[6], and Westinghouse Procedure RMF 8402[11],

From the load-time curve (Appendix A), the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined. Under some test conditions, a sharp drop in load indicative of fast fracture was observed. The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A).

IMPORTANT: Indication of " fast fracture" under stress tests.

Samples R53 and R54 (below) exhibit “fast fracture” characteristics at between 200 °F and 250 °F, quite close to the DBTT limit of <270 °F. This indicates significant embrittlement, and no amount of recalculation of the predicted values should be used to change the evidence of the material stress tests.

Table 5-8 Instrumented Charpy Impact Test Results for the Diablo Canyon Unit 1 Correlation Monitor Metal Irradiated to a Fluence of 1.37 x 10^{19} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy E_D/A	Max. E_M/A	Prop. E_P/A	Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
R49	25	4	32	14	18	1781	0.11	1839	0.12	1822	0	59	60
R51	50	8	64	36	29	3483	0.15	3593	0.16	3588	0	116	118
R56	100	10	81	36	44	3355	0.15	3465	0.17	3450	0	112	114
R55	150	33	266	185	81	3247	0.15	4399	0.44	4394	472	108	127
R53	200	47	379	229	150	3129	0.15	4449	0.53	4301	998	104	126
R54	250	74	596	301	295	3157	0.15	4377	0.67	4185	2576	105	125
R50	300	114	919	304	615	2971	0.15	4298	0.68	n/a	n/a	99	121
R52	325	120	967	313	654	3101	0.19	4228	0.73	n/a	n/a	103	122

Table 5-8 from the Capsule V Report, 2003, Pg.5-13

It is not clear from the table below whether sample #W10 represents the limiting weld #3-442C but there is some likelihood that it does. This would also need to be confirmed by the DCISC review. <u>We would ask the DCISC to assess such matters in detail.</u>

The Inappropriateness of the NRC’s License Approval Waiver

It seems clear from contact with the librarian at the NRC Docket that the NRC has never had possession of the fluence calculations which PG&E uses to replace the discredited stress test data. This is very surprising. The NRC clearly had many open issues in their review of PG&E’s fracture toughness figures at the time the LRA (license renewal application) was dropped in June of 2016. It is almost as though there was staff turnover and that staff may have forgotten how many open questions remained in 2016. When you speak with PG&E’s regulatory compliance staff, they will confidently assert that “the NRC has approved their fracture toughness calculations” and further assert that “PG&E has the highest reputation in the industry for managing the DCPP facility”. When it appears that the NRC never actually reviewed the 2011 fluence calculations, and the last valid study that is public on the docket indicates that Unit 1 will reach fracture toughness limits by 2021, only unqualified approval should be taken as NRC approval. There is nothing in the correspondence exchange between PG&E and the NRC in the 2015 to June 2016 timeframe to indicate approval of the new fluence calculations and projected 54 EFPY of Unit 1. The status of these “approvals” being uncertain, and with no written record to the contrary, it is hard to imagine how the NRC has waived the usual 5-year minimum review period for the License Renewal Application that they will not receive until end of 2023. One must ask is it appropriate for NRC to allow both reactors to continue operation until fall of 2026 without further stress test data? Yet that is what they have apparently done.

There are a host of regulations that would seem to prohibit further operation of the plant in the absence of these calculations: 10 CFR 50.61, ASTM E185-70 and 82, GL88-11, RG1.190 and RG1.99 would all suggest that these are required and not optional. <u>It would be my hope that the Diablo Canyon Independent Safety Committee would act *independently* to REQUIRE compliance with the fracture toughness requirements regardless of an NRC liberal interpretation of regulatory requirements.</u> The DCISC should at least advise the CPUC, the Governor, and the legislature of your *independent* findings as required by SB846.

Table 5-6 from the Capsule V Report, 2003, Pg. 5-11

5-11

Table 5-6 Instrumented Charpy Impact Test Results for the Diablo Canyon Unit 1 Surveillance Weld Metal Irradiated to a Fluence of 1.37 x 10^{19} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy E_D/A	Max. E_M/A	Prop. E_P/A	Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
W11	25	11	89	47	42	3939	0.15	4227	0.18	4217	0	131	136
W13	100	23	185	74	112	3426	0.14	4473	0.23	4405	267	114	132
W12	150	36	290	202	89	3440	0.15	4431	0.46	4422	573	115	131
W9	200	37	298	198	100	3281	0.15	4395	0.46	4315	206	109	128
W10	225	52	419	202	217	3354	0.15	4443	0.46	4272	1309	112	130
W15	300	71	572	211	361	3276	0.15	4290	0.49	n/a	n/a	109	126
W14	325	60	483	191	293	3141	0.14	4051	0.47	n/a	n/a	105	120
W16	350	66	532	203	329	3138	0.15	4182	0.48	n/a	n/a	104	122

There is considerable human cost at stake, especially considering the human toll, projected medical costs, liabilities, and the fact that the $24B+ in property value within the 50-mile radius of the plant cannot be fully insured by the either the state or federal governments. PG&E is exempted from being required to carry liability insurance to cover the loss to the residents of SLO County, should there ever be an accident. The public, the entire county depends upon the confidence that regulations on fracture toughness will be followed to the letter of the law.

The DCISC Deserves Credibility

It has been our impression, as well as that of other stakeholders that the DCISC has been balanced, diligent and acting in good faith to the public interest. Therefore, the questions raised today are not intended to question the integrity of this distinguished committee. And it is our hope that some of the contradictions in the record of PG&E's submissions to the NRC are in fact new information for the DCISC that will facilitate your further analysis. Please take time to review the NRC and PG&E correspondence excerpts in the appendices to this public comment as we believe it will cast light on the extent to which there have been numerous contradictions in the calculations and license application updates. Although PG&E may be allowed to try new regulatory compliance strategies, it is in fact important that the math be credible, and there are several regulatory documents that address such criteria that would require careful review.

Adjustments in the Regulations to Adjust For Conservatism

It is repeated in the literature and docket filings in numerous places that new mathematical calculations have been developed (10 CFR50.61a adopted 2010) to correct for "conservatism" in the prior methodology defined by 10 CFR 50.61. That is a credible argument on its face. It makes sense that after gathering coupon analyses on reactors all over the country for two decades, (Generic Aging Lessons Learned, etc.), the NRC wanted to adjust its conservative fracture toughness compliance rules (under 50.61) and offer an optional, "less onerous" alternative (50.61a). However, despite PG&E's stated intention to meet fracture toughness requirements under the new less onerous 50.61a rule, they were apparently not able to meet the qualification criteria for this less stringent rule. By 2011, they were back to qualifying under 50.61 using the language in Reg Guide 1.99, Position 1.2 to qualify a new fluence calculation method, which they have assured the NRC will meet all the required regulatory compliance requirements: 10 CFR 50 Appendices G & H, 10 CFR 50.61, ASTM E185-82 and RG1.99 & 1.190. It is not clear that the NRC has confirmed in writing that they are accepting PG&E's justification for the significant shifts in the data. There is nothing on the NRC docket to indicate that they have approved the new calculation methods without qualification, both because by 2016 there are still open concerns about the nozzle shell welds and because the NRC has never received the fluence calculation reports(WCAP-17299 & 17315).

The 2011 Revisions to the Unit 1 Fracture Toughness Assessment

The current status of the PG&E fracture toughness assessment was articulated in their annual update to the 2009 License Renewal Agreement dated December 21, 2011*[19] on the NRC docket (DCL-11-136). The new calculations yield results that make Unit 1 reactor compliant for operation through a total of 60 years (54 EFPY). In an effort to respond to the NRC's requests for clarification, PG&E's compliance strategy evolved from trying to invalidate the 2003 stress test data and complying under 50.61a, to using RG1.99, Position 1.2 and then adopting data from a similar reactor at Palisades. The NRC however, asked for clarification that the substituted data was both metallurgically similar to the materials in question, and that the general operational conditions of the "sister" plant were similar.*[20](See RAI 4.2.2-4, Appendix H) PG&E has replied that the conditions at Palisades are similar, but there is apparently no report on the NRC docket verifying that the chemistry factors metallurgical impurities) of the materials are the same and there is no independent confirmation that the sister plants have been operated in the same manner. All that is in the record are generalized assurances in a one-sentence response. (See Appendix K for excerpts from DCL-115-121). One nuclear expert and former inspector that was interviewed for this report, Michael Peck, stated that he knew of significant operational differences between the DCPP and the Palisades facilities.*[21]

Reasonable Questions

1) Why if there was any doubt about the accuracy credibility of the 2003 coupon analysis and projected 2021 expiration date, has PG&E requested delay of additional metal stress tests to confirm that the reactor pressure vessel is actually safe to operate?
2) Why has PG&E made several attempts to invalidate their own 2003 material stress test data?
3) Why in 2015 did PG&E request and NRC granted a waiver of inspection requirements for the critical welds in question?*[22]
4) Why is PG&E reversing its prior commitment to in-service inspection programs of the reactor vessels by requesting to discontinue Ultrasonic Testing (UT)?
5) Has the NRC fully assessed the risks to the public of granting a waiver of the usual 5-year LRA review period?
6) Is it appropriate that PG&E be granted even further delays in withdrawing Capsule B from Unit 1 to confirm the actual fracture toughness of the RV materials?
7) The DCISC has suggested that there are now new test methods that can allow an accurate retest of the Capsule Y and V samples, even if they have been tested to failure. How quickly can this testing be implemented and can test results be obtained prior to the scheduled outage in October of 2023?

The Potential for Reactor Vessel Failure by Pressurized Thermal Shock

The DCISC acknowledges that: "One event with the potential to generate sufficiently high stresses involves the injection of cold ECCS [emergency core coolant system] water under pressurized conditions, causing high, localized thermal stresses in the RPV. Because no large vessel is entirely free of minor cracks and other flaws, the phenomenon of concern would be that under high stresses one of these existing, small cracks might grow to become a major flaw or breach in the vessel's integrity".*[23] The science around such failures is well known and can be precisely calculated: "The temperature at which this transition (failure) occurs, in a narrow range of only 10-20 ^{o}F is known as the 'Reference Temperature for nil-ductility transition' (called RT_{NDT})."*[24] In the Charpy Impact Test parlance this temperature is also known as the Ductile-Brittle Transition Temperature (DBTT). Given the range of certainty of the 2003 material stress testing data, it is not hard to predict mathematically that failures will occur within a narrow range of temperatures. Given that the 2003 tests revealed limiting weld 3-442C failing at 280.4 °F, just above the DBTT limit of <270 °F, these results would indicate that we are at the limit of allowable fracture toughness now on Unit 1. This implicates quite significantly public safety issues.

A reasonable compromise would be to allow Unit 2 to continue operating for 7 to 10 years and shut Unit 1 down immediately if PG&E is out of compliance with ASTM E185-70 & 82. Allowing Unit 2 to continue operating for longer than 5 years would extend the transition to other generation alternatives such as solar, while also freeing up grid transmission capacity at DCPP to accommodate offshore wind development in the coming 5 to 10-year time frame.

It is well understood that the metallurgical properties of a pressurized water reactor and particularly welds at the cold water ECCS ports and extended beltline region are of critical importance to safe reactor operation. According to Sam Miranda, a nuclear engineer with decades of experience, the ECCS port welds are most likely to be subject to the highest localized stresses during a loss-of-coolant accident specifically because of the high temperature differential leading to localized stresses on RPV welds.*[25] According to Miranda, the Nuclear Regulatory Commission (NRC) does not require a failure modes scenario analysis (called FMEA or FMCEA) for multiple breaks in piping in the primary coolant system as may be induced by a design basis event such as a severe earthquake. Consequently, according to Sam Miranda, the ECCS system is not designed to control core temperatures under scenarios wherein there are multiple large cracks, and a more rapid loss of coolant poses a more significant risk of "pressurized thermal shock" (PTS) as valves close to contain the leak and the ECCS attempts to replace the lost reactor coolant. Additionally, a more embrittled reactor vessel is more likely to see additional stresses in a rapid shut-down scenario, so the larger the leak, the more ECCS coolant that is introduced, and the more stress that is imposed on the reactor core vessel increasing the risk of a RPV failure. The DCISC's 2011 "Evaluation of Pressurized Thermal Shock…" provides an excellent explanation of the safety concerns around such failure mode scenarios, and the detail in the report is of great value to public education on this important operational risk.

The reactor vessel is designed to withstand normal operating stresses, and this can be verified through calculations that during normal operating pressures, the vessel and its welds are sufficiently ductile (nonbrittle) because of high operating temperatures (approximately 600 °F). It is well understood that if there is any break or leak in a primary cooling system pipe, or any loss-of-coolant- accident (LOCA) such as a valve failure, monitoring systems flag the leak and the ECCS (Emergency Core Coolant System) automatically replenishes coolant that is stored at a relatively low temperature (outside ambient temperature typically 50° to 80 °F). The ECCS is designed to provide sufficient coolant to prevent fuel damage if any one of the nozzles or pipes leading to the reactor vessel breaks, it does not appear that the NRC requires that the ECCS be designed to provide sufficient coolant if several pipes break at once, if sufficient stress is induced by a "design basis event" such as an earthquake. The scenarios or "failure modes" through which the RPV may fail are centered around pressurized thermal shock due to a loss-of-coolant event, but failure mode scenarios that PG&E is required by the NRC to evaluate only consider those that fit within the designed parameters of the ECCS system. Multiple breaks, are not considered in the FMCEAs.

The question is one of risk analysis. Should failure modes and critical effects analyses (FMCEAs) consider the possibility of a more rapid loss-of-coolant-accident combined with the known embrittlement and the potential for pressurized thermal shock leading to a major rupture of the reactor vessel? To add to the scope of FMCEA variables, there are five leak detection systems at Diablo Canyon that are supposed to detect a 1-gallon leak per hour to confirm with the rules under the "Leak before Break" standards. In 2009 there may have been a system failure in which a leak occurred and all five of the leak detection systems were malfunctioning. (See Appendix O, 2009 LER). Under the "Leak before Break" guidelines, many cross supports were removed from large and small piping in order to give a longer response time to localized leaks as they slowly emerged. The redundancy of five leak detection systems is intended to provide a measure of safety, and there are strict NRC guidelines for shutting the plant down if a minimum redundancy is not provided.

Although the DCISC acknowledges that "the major concern is that an earthquake could initiate a cold-water safety injection leading to pressurized thermal shock",*[26] "[T]the sequences of greatest concern were found to involve medium diameter and large pipe diameter primary (cooling system loop) pipe breaks and stuck-open primary side valves that later reclose. For the pipe breaks, the fast cooling rates in some scenarios, combined with relatively low temperatures in the reactor…(arising from rapid depressurization and emergency injection of low-temperature make-up water…), combine to produce a possibly high-severity transient event (fracture of the reactor vessel)."*[27]

However, despite acknowledging these risks, the DCISC's concludes: "The possibility that the RV [reactor vessel] itself might break (during a loss-of-coolant- accident induced by seismic G-forces) is considered to be so unlikely as to be beyond-design-basis-event (earthquake), for which the ECCS need not be designed." *[28]

It is important to ask if the public understands the complexity of the assumptions being made (and risk scenarios missed) when such assurances are offered? Is it possible to imagine more than one pipe breaking during a significant seismic event – especially if many cross supports have been removed since the original construction of the plant due to the "Leak-Before-Break" policy?

Similarly, if the materials testing performed in 2003 suggests the plant should have been shut down by 2021 due to embrittlement, has any agency run the numbers to show what kind of G-forces the embrittled and stress-corroded pipes can withstand? Are the G forces and scale of earthquake required to trigger such an event in fact much lower than currently assumed? Do the plant operators realize that their ECCS emergency systems are not designed for such contingencies? Do operators consider such possibilities when running drills in their emergency management simulations? When such questions are unanswered, it would seem that thorough ultrasonic inspections would be called for.

PG&E States the NRC has Approved Substitution of Test Data from Another Reactor

PG&E has confirmed in direct email correspondence from their regulatory team that they were able to obtain NRC approval to substitute metal coupon stress tests from another similar (but not metallurgically identical) Westinghouse reactor. This is remarkable because it is well-known that the Unit 1 reactor vessel had metallurgical flaws in both the plate and weld materials, the latter containing impurities of copper and nickel that compromised structural resilience. Furthermore, the Palisades plant that is regarded as a "sister" plant to DCPP Unit 1 does have seemingly identical operating conditions and is currently undergoing its own investigation of radiation embrittlement.

Excerpts from an April 2023 email sent to the co-author of this paper, Bruce Severence by Tom Jones, PG&E's Government Relations Director stated the following:

> *"**Have the DCPP PTS evaluations been independently reviewed?** Both the NRC and DCISC have conducted independent reviews of DCPP's PTS evaluations and agree the DCPP reactor vessels are within the NRC's limits for embrittlement. See the DCISC's evaluation **here**, Section 4.23.2 (page 323 of the PDF).*
>
> *The DCISC has previously reviewed the proprietary information [the "proprietary" Westinghouse Report, probably report #WCAP-17299-NP, which justifies reinterpretation of the 2003 coupon analysis #WCAP-15958]**. <u>The NRC regulation was revised while the initial license renewal application was under review to allow for the use of representative coupons from other plants to be used for reactor vessel integrity evaluations.</u> This revised methodology is an acceptable methodology for the entire nuclear industry and was applied for Diablo Canyon Power Plant and was reviewed and accepted by both the NRC and the DCISC. As has been stated in public meetings, PG&E is planning to remove and test the remaining Unit 1 reactor vessel coupon*[25B] as part of the new license renewal (LR) application. During the LR process, the NRC will further evaluate this issue."*[29]*
>
> **NOTE: The only "new" coupons or surveillance data since the last stress test in 2002-2003 are from other "sister" reactors, and this is not disclosed on the tables where this data is presented. NRC requires proof of similar operating conditions when such data is substituted.

(Emphasis added.)

For the record, it is not at all clear that the NRC has fully approved the substituted data from Palisades. NRC's last RAIs in 2015 requested confirmation from PG&E of similar alloy specifications and operating conditions. NRC even suggest other reactors that may be suitable, but the docket record does not suggest approval of the approach.

The 2011 DCISC "Evaluation of PTS…", seems to clearly state that its endorsement of extended operation approval is a "preliminary" conclusion and pending final approval by the NRC. It remains unclear that there is any document on the NRC docket that confirms its final endorsement of PG&E's fracture toughness assessment, and the paragraph in a 2015 NRC letter that PG&E sometimes quotes as an approval, is an affirmation of a calculation, but within the context that there are still questions and details to affirm. There seems to have been a number of open issues and unanswered questions at the time that PG&E withdrew the 2009 LRA in June of 2016. The DCISC should conduct a thorough review of the exchange of correspondence regarding the relicensing.

Although PG&E's March 2023 letter to the NRC indicates that PG&E will attempt to withdraw Capsule B from the Unit 1 reactor in October 2023, the letter also admits that PG&E has had trouble removing an access hatch in 2010 that prevented them from doing so, and they are asking permission to delay retrieval of Capsule B until spring of 2025 during the 25th refueling outage and only if they are unsuccessful in removing the hatch.*[30] This would delay stress test results and a credible fracture toughness assessment until late 2026. In the absence of more reliable data, Unit 1 may be allowed to operate up to five years past its fracture toughness limits without verification of safety margins. This is unacceptable and appears to be a direct violation of regulatory requirements.

A Matter of Moral and Safety Priorities

The DCISC and PG&E both acknowledge that the Unit 1 reactor vessel was critically flawed from the outset, so ALL discussion should reference the inherent metallurgical flaws in the Unit 1 RPV components and welds wherein they are more highly susceptible to radiological embrittlement. If PG&E is looking for ways to invalidate their own 2003 stress test data, clear guidance and interpretation of RG1.99 Criterion 3 should be offered by the NRC and the DCISC before such data is deemed "not credible". If there are hairs to split on regulatory interpretation, it is best to err on the side of safety.

While referring to the three criteria for applying a different fracture toughness standard under 10 CFR 50.61a, the DCISC report states: "The overall conclusion was that the threat of a PTS [pressurized thermal shock] scenario is small IF certain metallurgical parameters are maintained in the vessel and its welds."*[31] (emphasis added.) The operative word is "IF". Unit 1 is known to have metallurgical flaws and so less stringent rules for estimating fracture toughness should not apply. It appears that the less stringent rules of 10 CFR 50.61a could not be applied because of the known metallurgical flaws. The principle remains the same even if PG&E is not using the 10 CFR 50.61a rules to comply. Furthermore, the fact that the plant has documented metallurgical flaws should give pause to any operator attempting to relicense the reactor when the only "hard data" in hand, the 2003 Capsule V Report and PG&E cover letter (DCL-03-052) indicate that Unit 1 is already past its useful operational life.

PG&E has taken action to invalidate the 2003 coupon stress-test data on several fronts. In PG&E, letter to the NRC dating back to July 2002, PG&E requested NRC review and approval of PG&E's proposed development of a different Pressure-Temperature Limit Report (PTLR) methodology that would allow PG&E to calculate "new pressure-temperature and LTOP limits without prior staff approval". *[32] This request was made roughly 60 days after the Capsule V material surveillance stress test was begun in 2002.

Appendix H of 10 CFR 50 "requires light-water nuclear power reactor licensees to have a reactor vessel (RV) material surveillance program to monitor changes in the fracture toughness properties of the RV materials adjacent to the reactor core." Standards for conducting material surveillance tests (stress tests of fracture toughness on material samples are also guided by the American Society for Testing and Materials International (ASTM) E185-82. According to these standards, surveillance program design must meet the requirements of the edition of ASTM E 185 that is current on the day the reactor vessel was purchased.*[33] These rules would seem to inhibit the NRC from granting PG&E the ability to develop its own compliance methodology, and it would certainly seem to limit the ability of the NRC to grant a waiver on these requirements.

In the following excerpt, PG&E states that the Upper Shelf Energy (a measure of fracture toughness) based on stress test data (Position 2.2) are most limiting, then they ***must*** be used regardless of the credibility of the surveillance data". There are ironies here. Reg Guide1.99 was used to invalidate the DBTT data) in favor of the USE calculation to avoid the problem of the limiting weld #3-442C not conforming to the 54 EFPY projections. Here the rule is acknowledged but applied inconsistently to reach the acceptable limits of fracture toughness. (See complete text of excerpt below in Appendix P*[34])

PG&E Response to RAI 4.2.3-1

PG&E amends LRA, Section 4.2.3, as shown in Enclosure 3, to state that in accordance with RG 1.99, Revision 2, the C_V USE data from Unit 1 surveillance Capsule V were determined not to be credible for determination of ΔRT_{NDT}, but were credible for determining the USE projections for Unit 1 RPV components represented in the DCPP RPV surveillance program for Unit 1.

RG 1.99, Revision 2 defines two methods that can be used to predict the decrease in USE due to irradiation. The method to be used depends on the availability of credible surveillance capsule data. For vessel beltline materials that are not in the surveillance program or are not credible, the Charpy USE (Position 1.2) is assumed to decrease as IMPORTANT: a function of fluence and copper content, as indicated in RG 1.99, Revision 2. PG&E indicates their read of RG1.99 to mean Position 1.2 can be used only if stress test data can be deemed "not credible". When two or more credible surveillance data sets become available from the reactor vessel, they may be used to determine the Charpy USE of the surveillance materials.

The surveillance data are then used in conjunction with Figure 2 of the RG to predict the decrease in USE (Position 2.2) of the reactor vessel materials due to irradiation. If the end-of-license and/or end-of-license extended USE values calculated using Position 2.2 are most limiting, then they must be used regardless of the credibility of the surveillance data. The above contradicts basis for invalidating the stress test data.

Unit 1 USE values were projected to 54 EFPY of operation using Position 1.2 results because they were more limiting than the Position 2.2 results. The surveillance data was most limiting under position 2.2 wherein it did not appear that weld #3-422C was the limiting component that would not comply with NDTT limits, but did under USE.Can PG&E pick & choose inconsistently?

A More Technical Review of Neutron Irradiation Damage

The issue at hand is the accumulation of neutron irradiation damage of the reactor pressure vessel (RPV or RV) of Diablo Canyon Unit 1 and whether the reactor's life can be safely extended for an additional 20 effective full-power years of operation (EFPY). For the lay person to fully understand the issues at stake, it is necessary to delve into a little nuclear engineering and nuclear chemistry and physics so that they become conversant with the technology. While this may seem to be excessive to some, we are of the opinion that only those who are technically informed on the technology can fully appreciate the issues at stake in any safety analysis of nuclear power reactors. We do this by first describing the reactor itself and Figure 1 (following page) provides a cut-away view of the reactor pressure vessel of a typical pressurized water reactor (PWR).

The component of principal interest is the active core length (the "beltline") of the reactor pressure vessel (RPV). It is this region of the RPV that is subjected to the most intense neutron irradiation of high energy neutrons (E = 14 MeV at the source, the fuel). In nuclear fission of $^{235}U_{92}$ in the fuel, the uranium atom's nucleus absorbs a thermalized (slow) neutron to form an excited, unstable species, $^{236}U_{92}$ The nomenclature used here is the U identifies the element, which is also given by the following subscript 92 (the number of protons in the nucleus) and the leading superscript 236 identifies the atomic mass (the sum of neutrons and protons in the nucleus) and hence the isotope. Natural uranium is composed of two isotopes, $^{235}U_{92}$ and $^{238}U_{92}$ with abundances of 0.7 and 99.3 %, respectively. This unstable entity, $^{236}U_{92}$, splits into roughly two equal fragments ($^{138}Ba_{56}$ and $^{96}Kr_{36}$) along with γ-photons (1-20 MeV), 2-3 neutrons, and considerable total energy ($E_{fiss.}$ = 202.5 MeV), as depicted in Figure 2. The fact that 2 – 3 neutrons are produced per fission event that requires only one thermalized neutron to initiate, allows for a chain reaction that produces sustained power. The reaction that produces three neutrons is shown in Reaction 1.

Formula (1) $$^{1}n_{0} + ^{235}U_{92} \rightarrow ^{141}Ba_{56} + ^{92}Kr_{36} + 3^{1}n_{0}$$

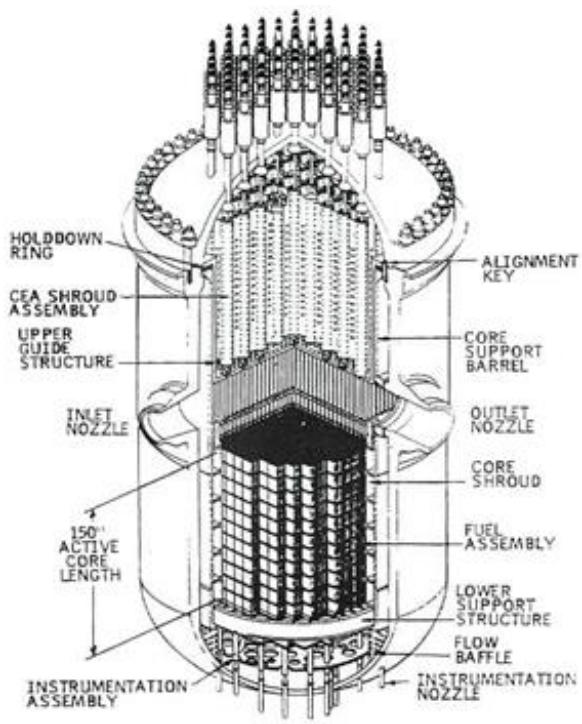


Figure 1: Cut-away schematic of the core of a typical pressurized water reactor (PWR).

where, for each species, the preceding superscript is the atomic mass (number of protons and neutrons in the atom's nucleus) and following subscript is the atomic number (number of protons in the nucleus), which identify the isotope and the element, respectively, as noted above. Reaction 1 is a typical nuclear transformation reaction in which new elements (Ba and Kr) are generated from an entirely different element (U) and a neutron. Nuclear reactions contrast chemical reactions where the same chemical species must appear on both sides of the reaction equation. The reaction expressed in Equation 1 above must mass balance and charge balance in the nuclei. Accordingly, mass balance requires that the sums of the leading superscripts on both sides of the reaction are the same (1 + 235 = 141 + 92 + 3 = 236). Likewise, for the atomic number (number of protons in a nucleus0, 0 + 92 = 56 + 36 = 92. This requires that a significant fraction of the thee fast (14 MeV) neutrons be "thermalized" by collision with a moderator that absorbs the neutron's energy without (ideally) neutron capture, thereby greatly reducing the neutron's energy (to 1 – 3 MeV) and velocity so that it may exist longer in the vicinity of the $^{235}U_{92}$ nucleus and hence greatly increase the probability of capture and induce fission (i.e., have a larger "capture cross section"). In the case of a PWR, the fuel is typically enriched in $^{235}U_{92}$ to 2.5% from a natural abundance of 0.7 % to increase the concentration of target nuclei, and the moderator is light water (H_2O). It is for this reason that BWRs and PWRs are often referred to as being "light water reactors" (LWRs).

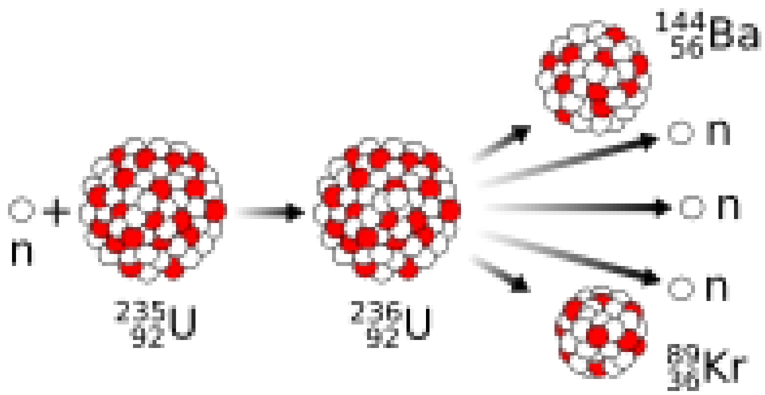


Figure 2: Schematic of the fissioning of $^{235}U_{92}$ by slow neutrons.

The reader will note from Figure 1 that between the fuel and the RPV there exists the stainless steel core shroud that has an attenuating effect on the energy of the neutrons released by the fissioning of $^{235}U_{92}$ in the enriched fuel (Figure 2). The thermal neutrons do not have sufficient energy (1-3 MeV) to penetrate the shroud and can be ignored in the following analysis, but they are essential for sustaining nuclear fission. Furthermore, the internal surface of the RPV is "clad" with an approximately 10 mm thick austenitic stainless steel liner (typically Type 316 SS) that also attenuates the energy of the fast neutrons before they encounter the RPV low alloy steel. On the other hand, the high energy ("fast") neutrons (14 MeV, Figure 3) readily penetrate the shroud, the coolant, and the stainless-steel cladding that exists on the inner surface of the RPV and a fraction of the high energy neutrons emitted by the fuel enter the low alloy steel (Typically A533) that is used to fabricate the RPV. It is these neutrons that result in the radiation damage that is the subject of this analysis, (Graph on following page).

The term "fluence" is used extensively in this report and the simplistic definition is that it is the product of the flux and time with the energy of the incident neutrons being specified. Thus, the fluence becomes $f = \varphi x t$ n/cm^2, where φ $(E > 1MeV)$ is the flux (n/cm^2.s) and t is the time (s). Noting that 1 EFPY = 3.15x10^7 s, the fluence becomes $f = \varphi x EFPY x 3.15\text{x}10^7$ n/cm^2. However, the neutron flux is time dependent, varying with the power level of the reactor, fuel burnup, fuel type and composition (enrichment), and other factors so that any accurate calculation of the fluence experienced after a given EFPYs must be redefined as:

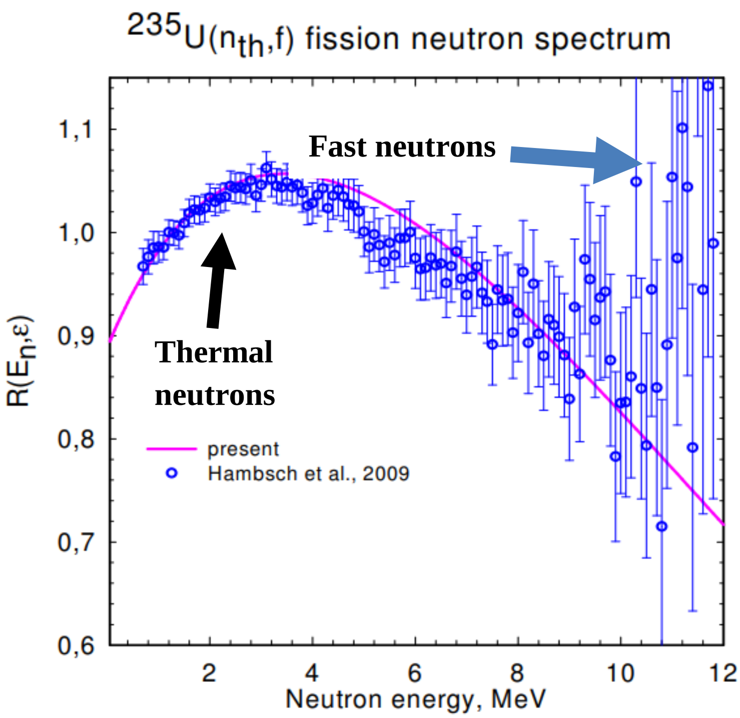
Formula (2) $$f(y\ EFPs) = 3.15\text{x}10^7 \int_0^y \varphi(t, E > 1\ MeV)\, dt$$

where the integration is best carried out piecewise over the operating history of the plant. The required data are generally available from plant records, but the integration can be laborious if all of the changes in plant operation are to be accurately captured. Although the details are not known to us, we presume that Equation (2) forms the basis of the PG&E/Westinghouse estimation of fluence.

As just noted, reactor pressure vessel walls are irradiated by fast neutrons that are responsible for material degradation. The incident, high energy neutrons (E < 14 MeV because of energy loss through the coolant, shroud, and liner) penetrate the RPV steel and their energy is dissipated by collision with metal atoms, displacing a small fraction of the metal atom from their usual lattice sites into the interstitial sites between the crystal lattice planes. The interstitial atoms then travel through the lattice and cause more atoms to become displaced, resulting in the creation of a "displacement cascade". Ultimately, this leads to defects, taking form as radiation damage including vacancies and interstitials which impede the movement of dislocations and result in a loss of ductility that is manifest as a hardening and embrittlement of the steel.

Over the course of the design life of a US pressure water reactor like Diablo Canyon (40 years), the wall of reactor pressure vessel (RPV) undergoes a displacement change of around 0.05 displacements per atom (dpa). One dpa signifies that every atom has been displaced once from its lattice site for a given fluence. Most displaced atoms, which form “Frenkel defects” comprising the displaced atom in an interstitial position and a vacancy from which the displaced atom came, recombine and the remaining point defects (Frenkel defects) affect the material's microstructure and mechanical properties, as described above. Recombination is a thermally activated process, meaning that the rate of recombination increases with increasing temperature.

Although poorly quantified, recombination effectively imposes an upper limit on the dpa when the rate of recombination matches the rate of formation of displaced atom. As noted above, point defects can impede



dislocations movement in a metal, making it more difficult for the material to plastically deform under a mechanical load, and resulting in the loss of ductility and hardening. As a result of hardening, the ductile-to-brittle transition temperature (DBTT) increases, which results in a material that was ductile at a given environmental temperature at a low fluence to become embrittled at a high fluence (total radiation damage over life of RPV) and the upper shelf energy (USE) decreases (Figure 4).

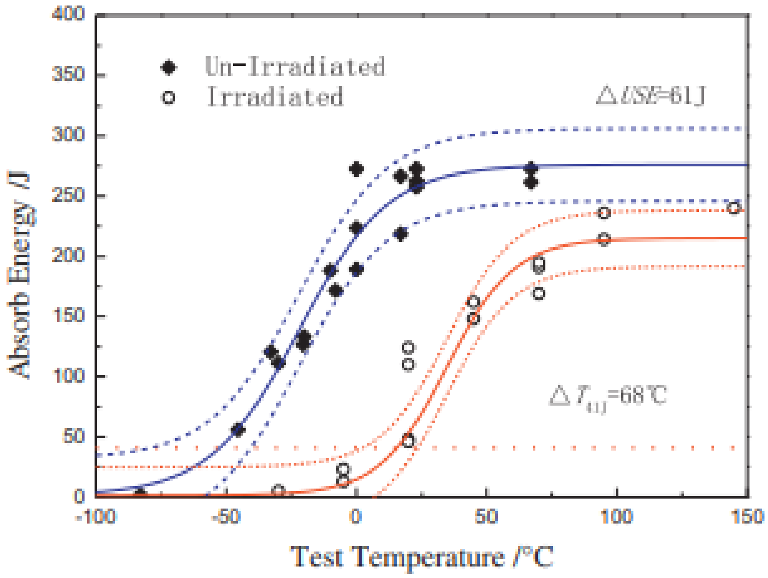


Figure 4: Effect of neutron irradiation on the Charpy impact test results for a fluence of 10^{20} n/cm^2 (E > 1 MeV) for A508-3 RPV steel. After Lin, et.al.

As noted above, the transition from the upper shelf fracture energy (ductile behavior) to the lower shelf fracture energy (embrittled microstructure and hence brittle fracture), as determined by the Charpy impact test method, is never sudden but occurs over a range of temperature and hence over a range of fluence as the fraction of the fracture surface displaying brittle facets (intergranular, brittle fracture) increases and that displaying ductile fracture facets (e.g., ductile tearing and micro void coalescence) decrease. In the case shown in Figure 4, the range extends over nearly 50 ^{o}C, which attests to the imprecision of the data. Also shown is that the DBTT shifts in the positive direction by about 50 ^{o}C for the stated fluence and the USE is reduced by 61 J.

At pressurized water reactor operating temperatures, neutron irradiation can also induce the formation of multiple new phases in a material that is at first single-phase, like austenitic stainless steel. Such phase formation is made possible by the diffusion of point defects, and the resulting redistribution of atoms in the metal into a more energetically favored configuration (a new phase). Also, neutron irradiation may induce segregation of alloying elements that can affect the material's corrosion resistance. For example, Cr, an element that is known to improve the corrosion resistance of Fe-Ni alloys (in stainless steels) by forming a stable chromic oxide, “passive layer” on the surface, may be depleted in regions adjacent to the grain boundaries due to the precipitation of chromium carbides on the boundaries in a phenomenon known as irradiation-induced sensitization (IIS) and hence result in irradiation-assisted stress corrosion cracking (IASCC) that has plagued the reactor internals, such as the core shroud welds in BWRs and the highly cold-worked Type 316 SS core shroud bolts in pressurized water reactors for decades.

Another mode of neutron irradiation degradation is the swelling of voids (or vacancy clusters) in the metal. Void swelling tends to increase with displacement dose for Ti-modified Type 316, an austenitic stainless steel containing Ti, Mo, Cr, and Ni. The voids are commonly found to contain helium, $^{4}He_2$ that may originate from a (n,α) reaction involving some species in the steel or from the coolant (see below). Another mode of neutron irradiation degradation is the swelling of voids (or vacancy clusters) in the metal. Void swelling tends to increase with displacement dose for Ti-modified Type 316, an austenitic stainless steel containing Ti, Mo, Cr, and Ni.

The above is the classical metallurgical/mechanical view of radiation induced damage in materials in reactor service, but the story is more complicated than that. Thus, a lot of work over the past couple of decades suggests that the primary coolant plays an important role and should be included in any comprehensive description of the embrittlement and failure process. To do so, it is necessary to describe the primary coolant circuit (PCC) of a typical pressurized water reactor. A schematic of a typical pressurized water reactor primary coolant circuit is displayed in Figure 5. This diagram displays two steam generators (SG1 and SG2) that are connected to the core in parallel. The two Diablo Canyon reactors each have four such loops. The PCC also has a pressurizer that has two functions. First, it maintains the pressure in the PCC to be above the vapor pressure of the coolant at the reactor operating temperature (typically 330 to 350 ^{o}C) and hence to prevent boiling (although nucleate boiling on the fuel occurs) and secondly it is used to maintain the desired hydrogen concentration in the coolant.

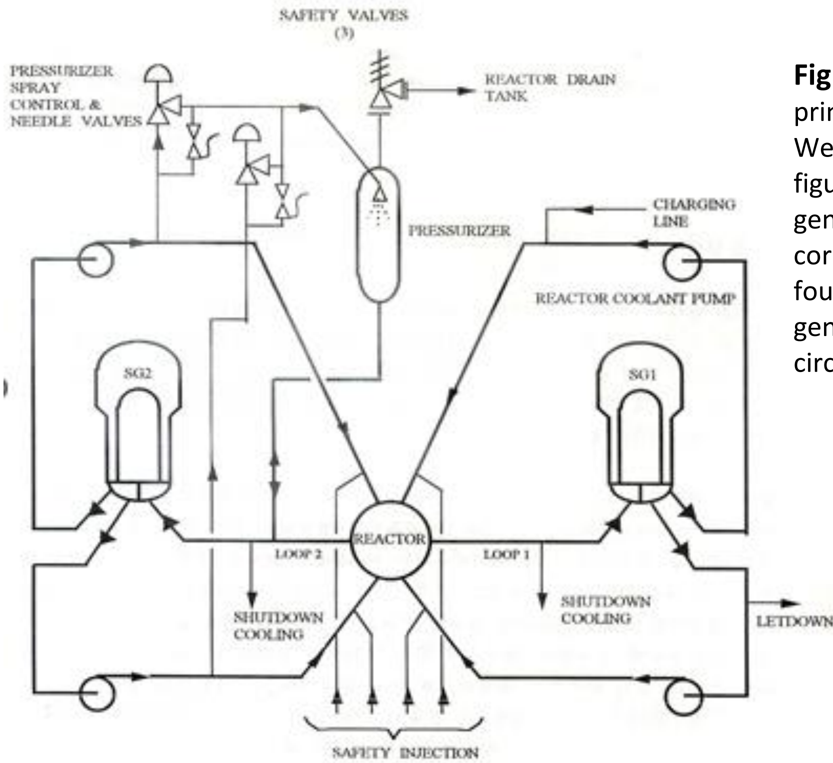


Figure 5: Schematic of the primary coolant circuit of a Westinghouse PWR. The figure shows two steam generators for the single core. Diablo Canyon has four parallel steam generator (SG1 and SG2) circuits.

Like all Westinghouse pressurized water reactors, the Diablo Canyon RPVs are lined with a thin layer of austenitic stainless steel (3-9 mm thick) for corrosion protection. The stainless-steel liner is in contact with the primary coolant that comprises a boric acid (H_3BO_3)/lithium hydroxide solution (LiOH) solution with an initial boron concentration of 1000 – 2000 ppm B and a lithium concentration of 4-1 ppm as shown in Table 1. The boric acid is present as a nuclear "shim" for reactivity control via the absorption of excess neutrons. The nuclear reaction involved is written as:

Formula (3) $\quad {}^{1}n_0 + {}^{10}B_5 \rightarrow {}^{7}Li_3 + {}^{4}He_2$

Thus, as burn-up of the nuclear fuel increases, the concentration of the boron decreases and that of the lithium from Reaction 2 increases but that would yield an undesirable pH (see below). Accordingly, it is necessary to control the Li concentration via ion exchange in the CVCS such that at the end of the fuel cycle the concentrations of boron and lithium are about 0 ppm and 0.4 ppm, respectively (Figure 6). The "burn-up" of boron, resulting in a progressively lowering of the boron concentration, is acceptable because with increasing burn-up of the fuel the fuel becomes less active due to the decrease in the concentration of fissile ${}^{235}U_{92}$. The pH of the coolant depends upon the ratio of the boron and lithium concentrations and hence the lithium and occasionally the boron concentrations must be controlled to achieve the desired trajectory of pH during burn-up within a fuel cycle (Figure 6). This trajectory is intended to minimize corrosion and, in particular, the transport of neutron-activated corrosion products in the primary coolant primarily to inhibit the deposition of the activated corrosion products in out of core areas such as the steam generators and the build-up of γ-radiation fields in the unshielded steam generator room.

Some of the "spare" neutrons depicted in Figure 2 are captured by the non-fissile isotope $^{238}U_{92}$ as described in the following reaction:

Formula (4) $$^{1}n_{0} + {}^{238}U_{92} \rightarrow {}^{239}U_{92} \rightarrow {}^{239}Np_{93} - \beta \rightarrow {}^{239}Pu_{94} - \beta$$

where -β represents the emission of an electron (beta particle) from the nucleus. The reader will note that with each emission of a beta particle, the atomic number increases by one corresponding to the conversion of a neutron in the nucleus into a proton (p^{+}) and an electron (e^{-}, beta particle), $^{1}n_{0} \rightarrow {}^{1}p_{1} + e^{-}$. Because the proton carries one positive charge, the reaction is charge balanced. The importance of Reaction (3) is that $^{239}Pu_{94}$ is fissioned by thermal neutrons and hence non- fissionable $^{238}U_{92}$ is converted into fissionable $^{239}Pu_{94.}$

Property	Value	Comment
Temperature	295°C - 330°C.	Typical
Pressure	150bar (2250psi).	Typical
Coolant Composition	4000–0ppm B as boric acid, 4–1ppm Li as lithium hydroxide, depending upon the burn-up of the fuel and the vendor.	Li-B trajectory over a typical fuel cycle is shown in Figure 38
Hydrogen Concentration	25 – 55cc(STP)/kg(H_2O)	Some non-commercial units operate with [H_2] as high as 70cc(STP)/kg(H_2O).
Core Channel Dose Rate		Typical
γ-Photon	3×10^{5}rad/s	
Neutron	6×10^{5}rad/s	
αParticles	3×10^{5}rad/s	
Coolant Mass Flow Rate	18,000kg/s	Typical

Table 1: Typical conditions that exist in the primary coolant of a Westinghouse pressurized water reactor.

This "breeder" aspect of a pressurized water reactor has important economic consequences in extending our uranium resources and it is estimated that at the end of a fuel cycle, about 40 % of the energy is produced by the fissioning of $^{239}Pu_{94}$.

As noted above, control of the pH is necessary for corrosion control and to minimize activity transport via corrosion products that are neutron-activated in the core and deposited in out-of-core components (e.g., the steam generators) that are not shielded against high energy γ-photons from isotopes like $^{60}Co_{27}$. The activation/decay process for $^{59}Co_{27}$ (an impurity in nickel alloys) may be written as:

Formula (4) $$^{59}Co_{27} + {}^{1}n_{0} \rightarrow {}^{60}Co_{27} \rightarrow {}^{60}Ni_{28} + e^{-} + \bar{\vartheta}_e + 2\gamma$$

(where $\bar{\vartheta}_e$ is an electron antineutrino.)

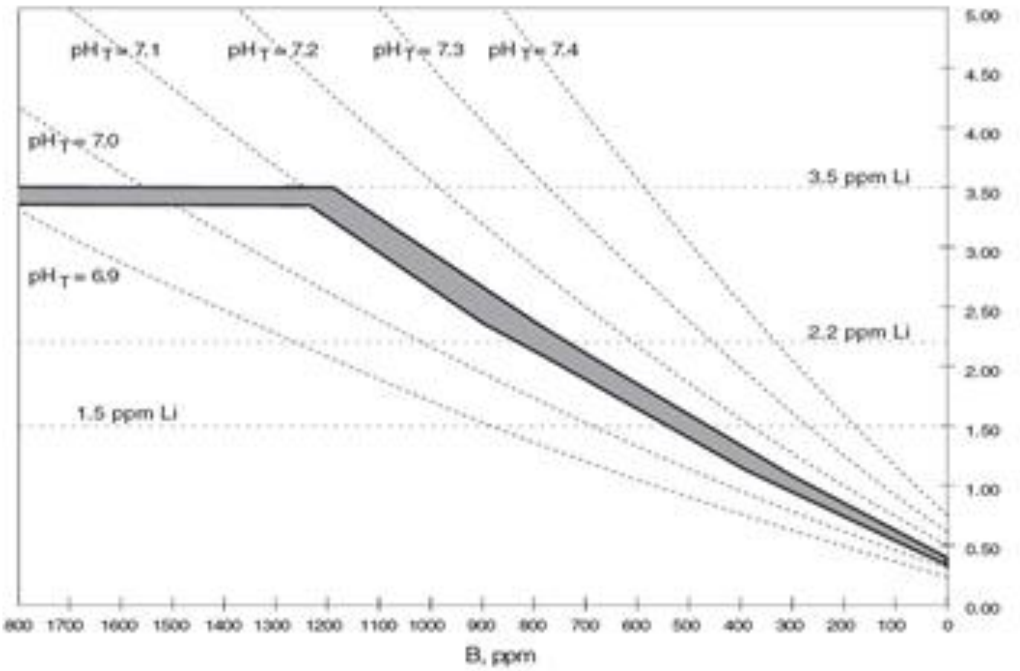


Figure 6: Desired trajectory of pH during fuel burn-up in a Westinghouse PWR.

where v_e is an "anti-electron" that is named a positron. The lithium and boron concentrations are controlled in the Chemical Volume Control System (CVCS) in the Reactor Water Clean-up System (RWCS) external to the core, such that the pH follows the trajectory represented by the dark gray area in Figure 6. Thus, initially, both [B] and [Li] are held constant at the indicated values until the pH at the reactor operating temperature In addition to boric acid and lithium hydroxide, the primary coolant typically contains 25 cc(STP)/kgH_2O (2.23 ppm or $2.11x10^{-3}$ molal) which is added to "suppress radiolysis" and prevent the formation of corrosive, oxidizing species such as oxygen, hydrogen peroxide, and the hydroxyl radical (HO). Besides the fact that this goal can be achieved at a far lower hydrogen concentration [5 cc(STP)/kgH_2O (0.446 ppm or $4.22x10^{-4}$ molal)], the molecular hydrogen (H_2) undergoes radiolysis to produce atomic hydrogen (H) as a part of the overall mechanism of the radiolysis of water*[35-37]. The [H] increases with the increase of [H_2] in the coolant and with increasing temperature and is a function of the pH. Some of the H penetrates the steel and recombines in voids (e.g., vacancy clusters) to pressurize the voids and provide additional driving force for void growth. Many of these voids nucleate on the grain boundaries and void growth eventually results in decohesion of the boundary and in brittle failure of the material. This hydrogen embrittlement phenomenon may be viewed as a "force multiplier" in the potential failure of RPVs but to our knowledge has not been considered by PG&E or by the NRC even though hydrogen embrittlement is well documented in corrosion science and has plagued other industries since their inception. For example, the phenomenon of "sulfide stress corrosion cracking" (SSCC) in the oil and gas industry is cause by hydrogen sulfide (H_2S) in the gas catalyzing the entry of atomic hydrogen into the steel where it recombines in voids particularly on the grain boundaries resulting in grain boundary decohesion and brittle fracture of the steel. The regions that commonly crack are the heat-affected zones (HAZs) adjacent to welds that have not or cannot be given an annealing heat treatment. The HAZs contain a very hard martensite phase that is particularly susceptible to brittle fracture. The parallelism between the SSCC case and reactor embrittlement (RE) is striking. In the SSCC case, the entry of atomic hydrogen into the steel from the coolant s due to H_2S inhibiting H recombination on the steel surface thereby forcing the atomic hydrogen to enter the substrate whereas in the RE case that function is provided for by the radiolysis of water. Thereafter, the mechanisms of failure are essentially identical. Again, drawing on the parallelism with SSCC, the most likely scenario where HE will be a significant factor in brittle fracture of radiation embrittled RPVs is during shut down under thermal shock conditions due to the injection of emergency cold water in response to a loss-of-coolant incident. Thus, at the operating temperature, the steel becomes saturated with atomic hydrogen that exists principally as interstitials in the lattice consistent with Sievert's law: $m_H = K^{1/2}p^{1/2}$where K is Sievert's constant, p is the hydrogen pressure in the environment, and m_H is the molal concentration of atomic hydrogen in the metal. The power of ½ arises from the fact that molecular hydrogen molecule must dissociate to form two atomic hydrogen atoms, H_2 = 2H, with $K = m_H^2/p$. According to Weiss[4], the solubility of hydrogen in various metals is a plotted in Figure 7. While the data only capture the PWR operating temperature (350 °C, 623K), it is evident that the solubility at this temperature is about an order in magnitude than at ambient (25 °C, shutdown).

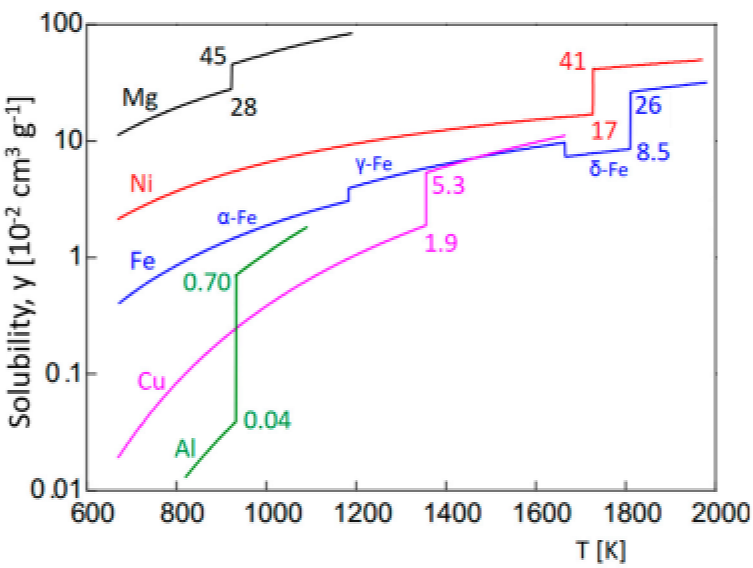


Figure 7: Solubility of hydrogen in various metals as a function of temperature, after Z. Weiss[*38].

Thus, the scenario in that at the operating temperature (623 K), the steel becomes saturated with hydrogen from the H_2 that exists in the primary coolant and that concentration is maintained as the system suddenly cools upon reactor shutdown under thermal shock conditions, the steel becomes supersaturate with the extent of supersaturation increasing as the temperature decreases. The steel responds by losing the excess hydrogen with the atomic hydrogen diffusing to nearest free surfaces, including the internal surfaces of voids, interfaces between the steel and precipitates and inclusions, grain boundaries, and the steel surfaces of the RPV. Because the voids, grain boundaries, and precipitate/inclusion interfaces are the closest free surfaces to any point in the steel except for a very thin layer at the RPV surfaces, the voids, grain boundaries, and precipitates/inclusions will receive the additional, supersaturation hydrogen, thereby greatly increasing the pressure in the defects. This is projected to provide additional driving force for delamination of the precipitates/inclusions from the steel matrix, void growth, and decohesion of the grain boundaries thereby enhancing the susceptibility of the already radiation-embrittled steel to brittle fracture. While this environmental effect is not (yet) encoded in RG1.99, it is obviously a factor that should be considered in any safety analysis or life extension of a reactor. Because of its "force multiplying" nature, it renders an already radiation-embrittled RPV even more susceptible to brittle fracture than would be the case if no hydrogen was present in the coolant.

Conclusions

PG&E's 2003 coupon analysis involved actual material stress testing on numerous samples that had been stored in Capsule V within the reactor such that they had a radiation exposure representative of the then 40-year projected service life (32EFPY) of the Unit 1 reactor. The 2003 material stress tests indicated some reactor vessel components and welds would be susceptible to failure by 2021 and some exhibited failure characteristics of "fast fracture"*[39]. The Capsule V Technical Report further identified weld #3-442C as a marginally compliant and "limiting" component with a projected 32 effective full-power years (32 EFPY, expiring approximately in Sept. 2021*[40]).

As early as July 2002, PG&E attempted to invalidate their own 2003 stress test data by citing of RG1.99, Rev2, Criterion 3, which freed them to develop new fracture toughness projections untethered by physical evidence. They then obtained NRC approval to develop PG&E's own Pressure-Temperature Limit Report methodology. The same RG1.99 criteria were applied to validate new USE calculations under the methodology outlined in RG1.99, Position 1.2 and PG&E applied the rules inconsistently to assert that "Unit 1 USE values were projected to 54EFPY (60-year period) using Position 1.2 results because they were more limiting than the Position 2.2 results" (Response to RAI 4.2.3-1 excerpted above). When invalidating the stress test data, PG&E asserted the opposite despite the DBTT values under Position 2.2 being far more limiting. Although PG&E correctly interpreted RG1.99 criteria in its 1993 Capsule Y Report, they interpreted it in an opposite fashion in order to have the 2003 stress test data and the Capsule V Report set aside. This inconsistency of rule application evidences an effort to sidestep the regulation to find a path for Unit 1 compliance.

Westinghouse and PG&E have collaborated to produce a report that is designated by the letters "NP" to signify "non-proprietary" (WCAP-17299-NP and WCAP 17315-NP, a PTS analysis)*[41], but which was not available to the public until July 3, 2023 when PG&E was finally required to produce the reports pursuant to a data request in the CPUC Rulemaking proceeding R.23-01-007. PG&E asserted that the new 2011 fluence calculation method justified their rewrite of the 2003 material stress test data, but it is still not clear how the documents justify changes in the physical data. A mathematical model that estimates fracture toughness cannot be as accurate as an actual stress test on a material sample to the point of failure. How then is it that we are debating invalidation of PG&E's material stress test data (very hard science) in favor of a much less certain calculation of fracture toughness? Based on the Westinghouse reports, PG&E has now confirmed that they have substituted material test data from another reactor vessel which may not have the same metallurgy and operating conditions as DCPP Unit 1. A former NRC inspector as well as an expert witness regarding the embrittlement of the Palisades reactor have asserted that the Palisades operating conditions differ significantly, a key concern to NRC staff as expressed in their requests for additional information on that reactor.

The fluence calculations seem to predict that radiation damage will decline or not continue to progress for the 20-year extended operation period. Given that the 2003 Capsule V Report had a high predicted-to-measured correlation, it appears the fluence calculations and projected fracture toughness were only very slightly conservative,, The new 2011 fluence calculation appears to reduce projected radiation damage by more than two orders of magnitude, and this scale of change is inconsistent with a "best fit curve" adjustment to Capsule V stress test results if we are to assume that data is in fact credible (See Table 5-10 of Capsule V Report*[42]). There are no known physical principles that would reverse radiological embrittlement, and the new fluence calculation appears to be beyond the limits of credibility. Investigation is warranted.

PG&E appears to admit that despite the large change in the fluence calculation, the intermediate nozzle shell welds in BOTH reactors do not meet fracture toughness limits through a 20-year extended operation period (Appendix F). The NRC's last correspondence to PG&E prior to the LRA withdrawal in February of 2016 suggests the NRC still questioned the nozzle shell welds' fracture toughness, among other issues. There doesn't appear to be a final approval of PG&E's fracture toughness and PTS assessment. The public deserves a reasonable, independent assessment of the actual risk of pressurized thermal shock scenarios, and the only means of providing it is through the immediate withdrawal of Capsule B to complete more credible stress tests as should have been previously delivered.

FOOTNOTES

*1a "Fracture Toughness" is used in this analysis to refer generally to methods of calculation, and a range of stress tests required by 10 CFR 50.61, 50.61a, ASTM E185-70 &82, Reg Guides RG1.99 and RG1.190, among others. Here "fracture toughness" is used as a term of art to clarify for general audiences the concept of gradual loss of it resulting in reactor vessel embrittlement. This term is not specifically intended to refer to K_{1C} or J_{1C} values as defined by fracture mechanics.

*1 Analysis of Capsule Y from The Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-13750, E. Terek, S.L. Anderson & A. Madeyski, Westinghouse Electric Corporation, July 1993

*2a &2b Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, Technical Report, ML031400342, WCAP-15958, January 2003, Pg.6-5 Also note: pg.5-1: "Under some test conditions, a sharp drop in load indicative of fast fracture was observed."

*3 Analysis of Capsule Y from The Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-13750, E. Terek, S.L. Anderson & A. Madeyski, Westinghouse Electric Corporation, July 1993, pg.1-2 & 1-3

*4a There is some mixed use of various mathematical terms such as RT_{NDT}, DBTT, NDTT, and RT_{PTS} in the literature and PG&E correspondence that merit clarification for the general public. According to the NRC description of 10 CFR 50.61 definitions, "RT_{PTS}" stands for Reference Temperature of Pressurized Thermal Shock, and means "the reference temperature, RT_{NDT}, evaluated for the EOL Fluence for each of the vessel beltline materials", using the procedures defined in 50.61. "The pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials." (https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0061.html) Licensees are required by 10 CFR 50.61 (and other regs) to demonstrate compliance with fracture toughness requirements. Specifically: "In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least three years before RT_{PTS} is projected to exceed the PTS screening criterion." As described in the PG&E Capsule Y Report (1993), Pg.3-1: "The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G to the ASME Code. The K_{IR} Curve is the lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature.

Allowable operating limits can then be determined using these allowable stress intensity factors." (See also footnote 4c)

*4b PG&E Cover Letter to the Diablo Canyon Unit 1 Reactor Vessel Material Surveillance Technical Report (Coupon Analysis), addressed to the NRC, May 13, 2003, Docket No. 50-275, Ref.# DCL-03-052, pg.2

*4c For the purposes of this analysis will refer to the "ductile-brittleness transition temperature" (DBTT) as the point at which a metal sample representing a weld or plate material in the reactor vessel transitions from more elastic behavior to the "fast fracture" or more brittle behavior. The maximum allowed by regulation for most welds and plates is defined by the <270°F threshold. Circumferential welds are allowed a threshold of <300°F. Fast fracture is defined by the characteristics of failure wherein metals lose elasticity (ductility) and there is a sudden drop in load typical of embrittled metal, resulting from radiation damage. This DBTT term aligns with the terminology that is most commonly used in the Charpy V-Notch material stress test methodology, reports and data tables. The "nil-ductility transition temperature" (NDTT) or "nil-ductility transient temperature" is often used interchangeably to refer to the DBTT threshold of <270°F, but it is more correct to refer to DBTT. Below the DBTT stress tests threshold of <270°F performed at 30ft.lbs. or 50ft. lbs. material samples are said to comply with minimum regulatory requirements. Any stress tests sample results with a DBTT above the <270°F threshold are said to have lost "ductility" at higher temperature, and would be said to have failed the stress test in accordance with regulatory requirements.

*5 Regulatory Guide 1.99, Revision 2, US Nuclear Regulatory Commission, Task ME 305-4, page 1

*6 DCL-11-136, PG&E correspondence to the NRC regarding the 2011 annual update to the 2009 License Renewal Application, ML12009A070, most relevant pages 78-88

*7 PG&E Letter to the NRC (DCL-12-124) "Annual Update to the Diablo Canyon Power Plant License Renewal Application and License Renewal Amendment Number 4, ML12356A179, dated Dec. 20, 2012, Pages 36 and 50 conflict on PTS related data.

*8 Neutron Embrittlement at Diablo Canyon Unit 1 Nuclear Reactor, A. Gundersen, Fairewinds Associates Inc., 2016, page 2

*9 NRC Request for Additional Information to PG&E, Set 39, (TAC NOS> ME2896 & ME2897), ML16011A365, February 2, 2016, Pgs. 7-8,(pdf.Pgs. 10-11)

*10 NRC Request for Additional Information RAI-Set 38 (TAC NOS. ME2896 &ME2897), ML15217A481, Sept. 24, 2015, Pg.7 as restated with PG&E responses in DCL-15-121 (October 21, 2015), Pages 17-18

*11 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011

*12 PG&E Cover Letter to the Diablo Canyon Unit 1 Reactor Vessel Material Surveillance Technical Report (Coupon Analysis), addressed to the NRC, May 13, 2003, Docket No. 50-275, Ref.# DCL-03-052, pg.2

*13 2009 License Renewal Application (LRA), DPR-80 and DPR-82, Pg. 4.2-8, pdf Pg. 948

*14 2009 LRA pg.4.2-4, pdf Pg 944

*15 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg. 2

*16 NRC Request for Additional Information RAI-Set 38 (TAC NOS. ME2896 &ME2897), ML15217A481, Sept. 24, 2015, Pgs.8-9

*17 Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, Technical Report, ML031400342, WCAP-15958, January 2003, pg.5-1: "Under some test conditions, a sharp drop in load indicative of fast fracture was observed."

*18 Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, Technical Report, ML031400342, WCAP-15958, January 2003, pg.5-11 & 5-13, Table 5-6 & 5-8

*19 DCL-11-136, PG&E correspondence to the NRC regarding the 2011 annual update to the 2009 License Renewal Application, ML12009A070, most relevant pages 78-88

*20 NRC Request for Additional Information RAI-Set 38 (TAC NOS. ME2896 &ME2897), ML15217A481, Sept. 24, 2015, Pg.7 as restated with PG&E responses in DCL-15-121 (October 21, 2015), Pages 8-10

*21 Phone conversation with Michael Peck, former NRC inspector (retired) on or about June 11, 2023

*22 Neutron Embrittlement at Diablo Canyon Unit 1 Nuclear Reactor, A. Gundersen, Fairewinds Associates Inc., 2016, page 4 “waived inspections”

*23 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg.6

*24 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg.6 (NDDT explained)

*25 Conversation with Dr Sam Miranda, Titles, NRC Risk Analysis Expert Witness, May 25, 2023

*26 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg. 1

*27 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg. 2

*28 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg. 6

*29 Emailed correspondence from PG&E government relations team, Tom Jones, dated April 13, 2023, see Appendix A below for full text.

*30 May 15, 2023 Letter from PG&E to the NRC, Revision to Unit 1 Reactor Vessel Surveillance Program Withdrawal Schedule, Docket No.50-275, OL-DPR-80, Ref. DCL-23-038, Requests approval of a new coupon withdrawal and test schedule, pg.5

*31 Diablo Canyon Independent Safety Committee Evaluation of Pressurized Thermal Shock and Seismic Interactions for Diablo Canyon Unit 1 Reactor, DCISC, February 15, 2011, pg. 5

*32 Letter DCL-02-079 “License Amendment Request 02-04, Revision of Technical Specification 5.6.6 – Reactor Coolant System Pressure and Temperature Limits Report (PTLR), dated July 31, 2002

*33 NRC Rule Doc Citation: 84 FR 12876, docket #NRC-2017-0151, doc#2019-06418 pg. 1

*34 PG&E Response to RAI 4.2.3-1, DCL-15-121, ML15294A437, October 21, 2015, Pg. 17-19

*35 D. D. Macdonald and M. Urquidi-Macdonald. " The Electrochemistry of Nuclear Reactor Coolant Circuits," *Encyclopedia of Electrochemistry*, A.J. Bard and M. Stratmann eds. Vol 5 Electrochemical Engineering, Edited by Digby D. Macdonald and Patrik Schmuki, Wiley-VCH Verlag GmbH & Co. KGaA, Weinheim, pp. 665-720, (2007).

*36 D. D. Macdonald, G. R. Engelhardt, and A Petrov, A Critical Review of Radiolysis Issues in Water-Cooled Fission and Fusion Reactors: Part I, Assessment of Radiolysis Models, Corrosion and Materials Degradation 3, 470-535 (2022).

*37 D. D. Macdonald, G. R. Engelhardt, A Critical Review of Radiolysis Issues in Water-Cooled Fission and Fusion Reactors: Part II, Prediction of Corrosion Damage in Operating Reactors, Corrosion and Materials Degradation 3 (2022), 694-758.

*38 Z. Weiss, Analysis of Hydrogen in Inorganic Materials and Coatings: A Critical Review, Hydrogen, 2, 225–245 (2021)

*39 Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, Technical Report, ML031400342, WCAP-15958, January 2003, Pg.5-1: "Under some test conditions, a sharp drop in load indicative of fast fracture was observed."

*40 PG&E Cover Letter to the Diablo Canyon Unit 1 Reactor Vessel Material Surveillance Technical Report (Coupon Analysis), addressed to the NRC, May 13, 2003, Docket No. 50-275, Ref.# DCL-03-052, pg.2

*41 Westinghouse Report WCAP-17299-NP "Fast Neutron Fluence Update for the Diablo Canyon Unit 1 and 2 Pressure Vessels", Revision 0, February 2011. NOTE: The title of an associated Westinghouse Report WCAP-17315-NP is not known.

*42 Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, Technical Report, ML031400342, WCAP-15958, January 2003, Pg.5-13, Table 5-10

*43 M. Griffiths, "Effect of Neutron Irradiation on the Mechanical Properties, Swelling and Creep of Austenitic Stainless Steels.", Materials 2021, 14, 2622. https:// doi.org/10.3390/ma14102622.