

**Diablo Canyon
Independent
Safety Committee**

**Public Meeting
October 20 and 21, 2026**

Avila Lighthouse Suites



Diablo Canyon Power Plant: State of the Plant Update

October 20, 2026



Justin Rogers - Diablo Canyon Station Director

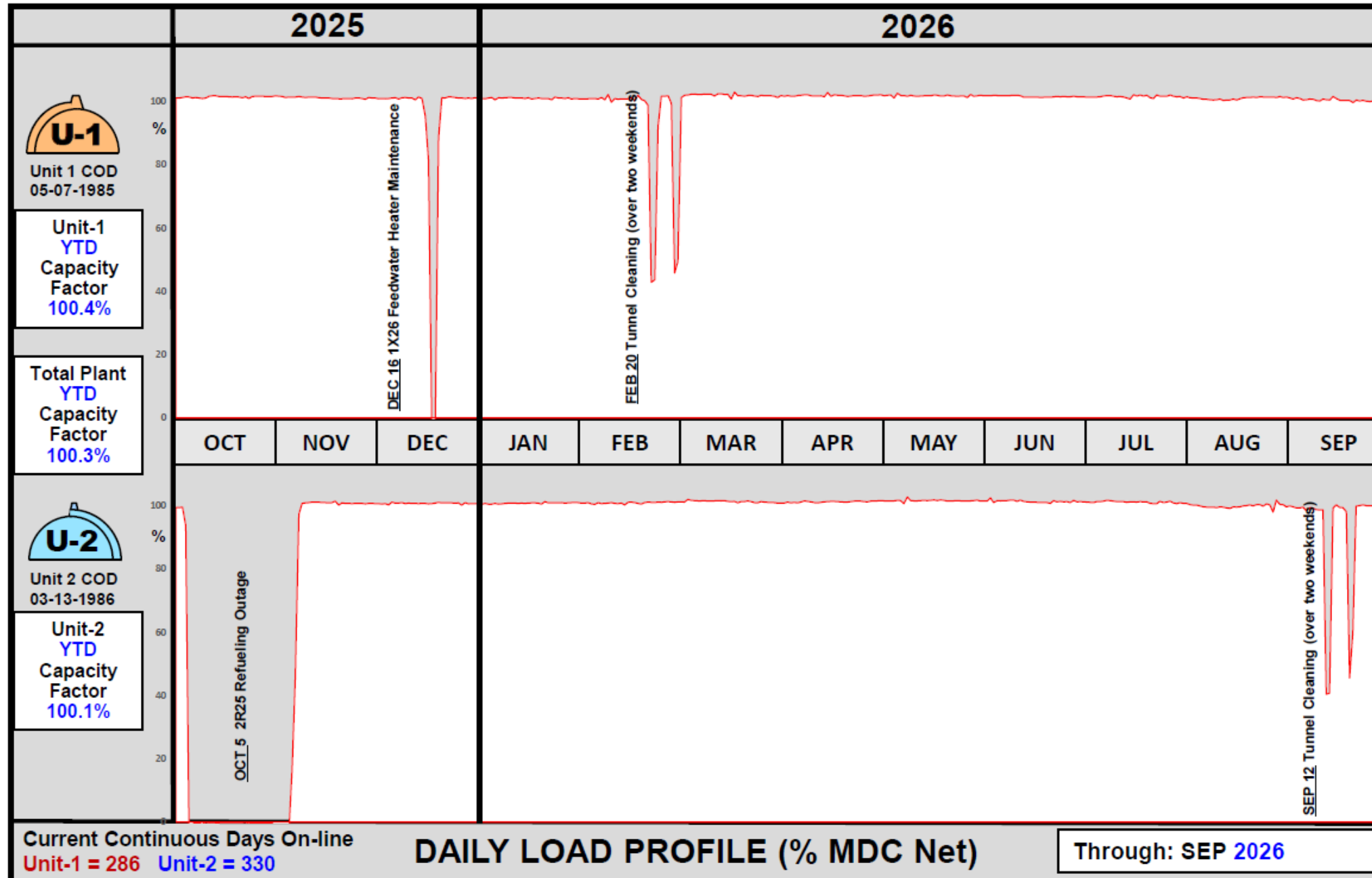
Plant Operation

- Unit 1 is in a planned Refueling Outage
- Unit 2 is safely operating at 100% power
- All Nuclear Regulatory Commission (NRC) Performance Indicators (PIs) are Green



Daily Load Profile: Past 12 Months

Daily Capacity Factor Power History Curves



Key Events - Recently Completed Activities

- **Startup Transformer (SUT) 2-2 feeder cable/bus maintenance performed in July**
- **Diesel Generator (DG) Maintenance Outage Window (MOW) August 2-16**
- **NRC Inspections:**
 - **Emergency Planning Exercise and Performance Indicator (PI) August 31-September 4**
 - **ISFSI aging management / license renewal inspection, August 17-20**
- **Unit 2 curtailment to 50% power for tunnel cleaning the weekends of Sept 18-20 and Sept 25-27 (each weekend focused on one of two circulating water tunnels for Unit 2)**

Key Events - Upcoming Station Activities

- **Unit 1 Refueling Outage continues**
- **NRC Inspections:**
 - **Radiation Protection, October 19-23**
 - **Inservice Inspection (ISI), October 19-30**
 - **Physical Security, December 7-10**

Recent Industry Awards

Year	Award	Recipient
2025	NEI Top Innovative Practice (TIP) Award	DCPP Innovation Team (AI Integration)
2025	NEI Top Innovative Practice (TIP) Award	DCPP Strategic Initiatives Team (Communications & Outreach)
2026	Richard A. Clarke Environmental Leadership Award	DCPP Environmental Services Team
2026	PG&E Progress Makers Recognition	Various DCPP Engineers
2026	NEI Women In Nuclear Best Overall Medium Chapter	PG&E Women's Chapter

Leadership Changes

- **Jordan Tyman, currently Director Risk & Compliance, transitioning to Senior Director Enterprise Compliance & Governance on Nov 1, 2026**
 - **Risk and Compliance will be reporting to Philippe Soenen, Director of Strategic Initiatives, PI & OR**



Diablo Canyon Power Plant: State of the Plant Update

Questions?

Justin Rogers
Diablo Canyon Power Plant

Diablo Canyon Power Plant Regulatory Performance Update

Jordan Tyman

Sr. Director, Enterprise Compliance & Governance



July 2026 – October 2026

- DCPP continues to provide safe, clean, and reliable electricity to Californians
- Since the last update to the DCISC, the NRC oversight of DCPP included approximately ~1,800 hours of inspection time
- No violations received since the last DCISC update
- DCPP remains in the highest performance category for all NRC Performance Indicators

DCPP remains in the highest performance category

- In addition to performing inspections, the NRC also evaluates every nuclear plant using performance indicators
- NRC gauges licensee performance using 16 separate indicators related to various aspects of plant operation
- DCPP remains in the highest performance category of the NRC's Reactor Oversight Process Action Matrix
 - No inspection findings of more than very low safety significance since the last DCISC update
 - Performance indicators within the expected range (i.e., green)



Regulatory Update



July 2026 – October 2026

NRC Violations

- DCCP received no violations during this period



Regulatory Update



October 2025 – January 2026

Licensee Event Reports (LER)

- There were no LERs issued during the reporting period



Upcoming Inspections



Upcoming NRC Inspections

Inservice Inspection – October 19 – 30, 2026

Radiation Protection – October 19 – 23, 2026

License Amendments

- DCPP received NRC approval of one License Amendment Requests (LARs) since the last update to the DCISC:
 - Utilize ADOPT™ Fuel for Improved Fuel Performance
 - 50.69 Enhanced Passive Categorization
 - NEI 99-1 Rev 7 Emergency Action Level (EAL)
- DCPP has submitted several LARs, which are with NRC for review and approval (e.g., E-Plan rebaseline to NUREG-0654 Rev. 2, etc.)
- DCPP plans to submit additional LARs over the next few months consistent with industry best practices (e.g., standard industry LARs, (e.g., TSTFs, etc.) and UHS Resiliency

DCPP Regulatory Performance

- DCPP remains in the highest NRC regulatory performance category
- DCPP continues to provide safe, clean, and reliable electricity to Californians



Thank You

Jordan Tyman

Sr. Director, Enterprise Compliance & Governance



Evaluated Exercise Performance Summary

Jordan Tyman

Sr. Director, Enterprise Compliance & Governance



Public and Employee Safety

- PG&E Emergency Plan is designed to protect the health and safety of the public
- Bi-Annual NRC Evaluated Exercise September 2, 2026
- Observed by NRC and FEMA

Effectively Implemented Emergency Plan

- All risk significant and critical actions were performed effectively
- No weaknesses were identified
- Effective coordination between PG&E and State and County responders
- DCPP received no violations during this period

Effectively Implemented Emergency Plan

- Emergency Response Organization (ERO) personnel demonstrated effective implementation of the Emergency Plan, demonstrating the ability to protect public health and safety
- All gaps entered in Corrective Action Program (CAP) to drive continuous improvement



Performance Indicator Evaluation



Classifications	2 out of 2 Timely & Accurate
Notifications	2 out of 2 Timely & Accurate
PARs	1 out of 1 Timely & Accurate
PAR Notifications	1 out of 1 Timely & Accurate
Overall Results	6 out of 6 Timely & Accurate

DCPP Regulatory Performance

- DCPP remains in the highest NRC regulatory performance category
- DCPP response during the exercise was successful at demonstrating the station's ability to prioritize actions needed to assure public health and safety
- DCPP received no violations during this period



Thank You

Jordan Tyman
Sr. Director, Enterprise Compliance &
Governance



DCPP Probabilistic Risk Assessment (PRA) Program Overview

Jordan Tyman

**Sr. Director, Enterprise Compliance &
Governance**



- PRA Overview
- U.S. Nuclear PRA History
- Diablo Canyon Power Plant (DCPP) PRA History
- Current Applications of PRA at DCPP
- The Future of PRA at DCPP – Applications and Models

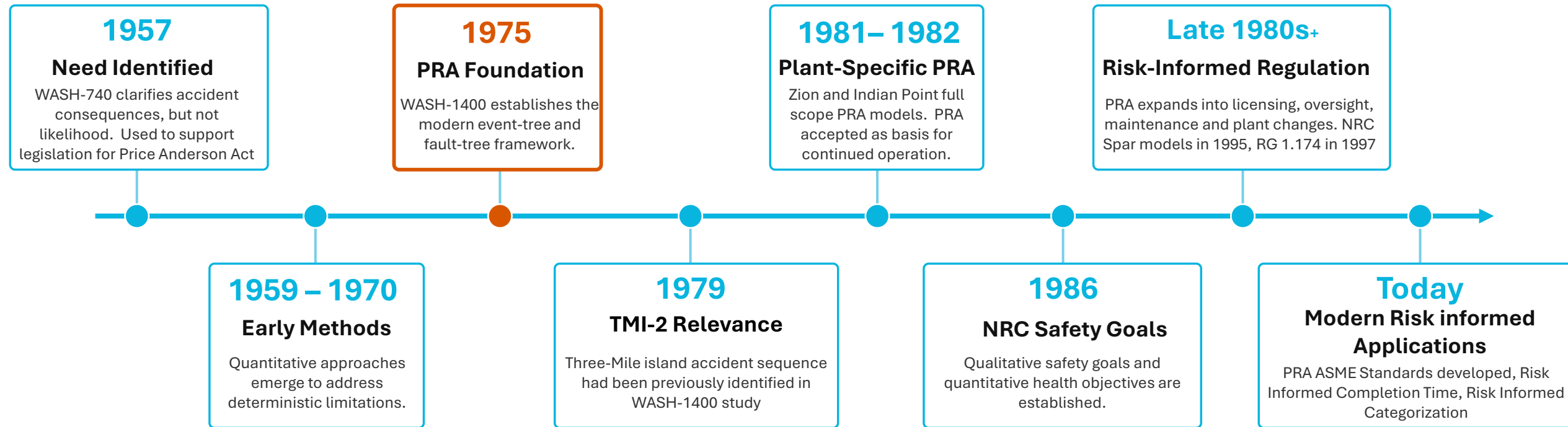
- PRA is an analytical tool that can:
 - Identify possible scenarios (Accident analysis, Industry experience)
 - Estimate likelihood of occurrence
 - Estimate consequences (Impacts to frontline and support systems)

- Provides specific technical analysis that systematically answers the risk triplet:
 - What can happen (i.e., normal conditions and/or off-normal conditions)? (scenario)
 - How likely is it to occur? (probability, frequency)
 - What will be the outcome? (consequences)

Introduction

- The PRA models are used to estimate the potential for an event to cause Core Damage: Core Damage Frequency (CDF)
- The models include contributions from:
 - Internal events
 - Fire
 - Flood
 - Seismic
- DCPD PRA models have been independently reviewed by industry peers to meet ASME standards accepted by the NRC
- DCPD PRA models are continuously updated incorporating design changes, procedure changes and new data to ensure the PRA matches the as-built facility

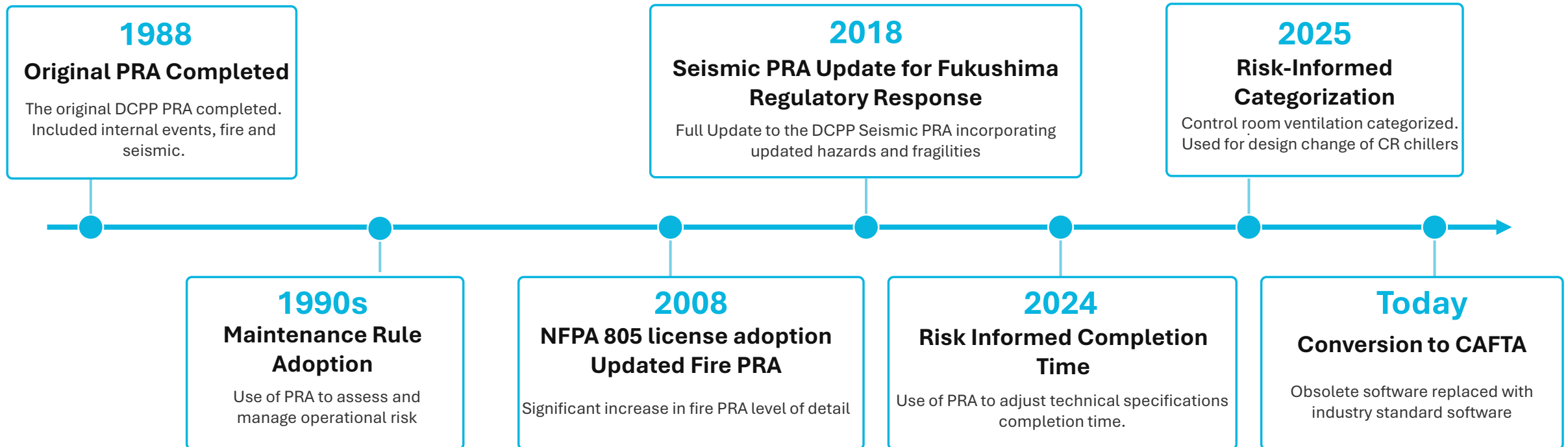
Timeline of Key Milestones



PRA shifted nuclear safety toward risk-informed decision-making

What can go wrong? • How likely is it? • What are the consequences? • Where should risk be reduced?

Timeline of Key Milestones



PRA shifted nuclear safety toward risk-informed decision-making

What can go wrong? • How likely is it? • What are the consequences? • Where should risk be reduced?

- Online risk assessments – PRA is responsible for maintaining the model and software that Operations and Work Control uses for these assessments
- Risk-Informed Completion Time (RICT) – PRA supports model development and use for calculation of risk informed completion times
- Risk-Informed Surveillance Frequency Control Program – PRA supports testing interval increases
- 50.69 – Risk informed categorization and treatment of Structures, Systems, and Components (SSCs) – License Amendment Request (LAR) approval received in December of 2024

Future Models and Applications

CAFTA Model Conversion. DCPP Completed a conversion of the existing Riskman model to CAFTA. Benefits of CAFTA conversion include:

- Common framework with industry
- Software compatibility and support
- Faster quantification time



Thank You

Jordan Tyman

Sr. Director, Enterprise Compliance & Governance





Per Peterson, Member, with Richard McWhorter, Consultant

- 1. Dropped Control Rod Event Corrective Actions**
- 2. Major Project Planning**
- 3. Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring**
- 4. Observe Leadership Alignment Meeting**
- 5. Condensate and Feedwater System Health**
- 6. Station Update**
- 7. Reactor Vessel Embrittlement Sample Capsule B Report**
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Dropped Control Rod Event Corrective Actions

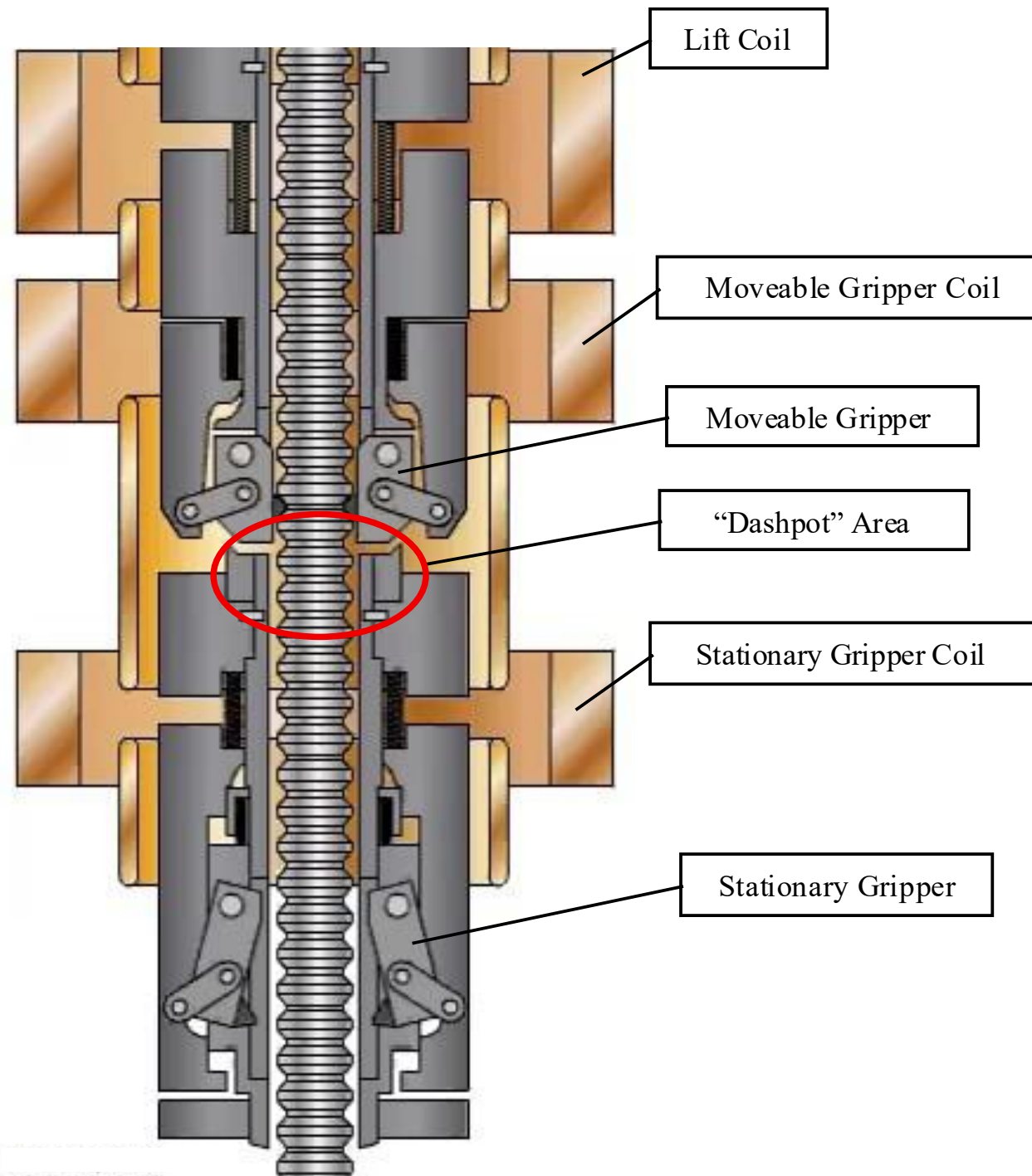
- **August 2024; Control Bank D Rod P10 Fell Into Core During Power Reduction**
- **Vendor Guidance Followed to Troubleshoot Rod Control System**
- **Electrical Issues Eliminated as Possible Cause; Rod Withdrawn 18 Hours Later**
- **Concluded Likely Due to Corrosion Product Buildup Around Movable Gripper**
- **Industry Issue; Guidance Was in Development**
- **Did Not Affect Safety Function of Rods**
- **Recommendation for Follow-up Opened (D11/24FF-3.13)**



Dropped Control Rod Event Corrective Actions (continued)

- **May 2025; Similar Event – Slippage of F02 119 to 90 Steps During Inward Move**
- **Rod Control System Troubleshooting Eliminated Electrical Issues**
- **Since 2024/2025, DCPD Completed Extensive Reviews of Industry Information**
- **Data Continued to Point to Small Amounts of Corrosion Buildup in “Dashpot” Region of Control Rod Drive Mechanism (CRDM)**
- **Investigations Included Disassembly of a CRDM at a Sister Plant**
- **Industry Data for Multiple Events Failed to Find any Correlation to Contributing Factors Except CRDMs on Periphery of Core More Susceptible**
- **Industry and DCPD Data Showed No Correlation to Surveillance Frequency Changes**

Control Rod Drive Mechanism Arrangement





Dropped Control Rod Event Corrective Actions (continued)

- **Most Corrective Actions and Prudent Actions Complete**
- **Most Significant Change – New Surveillance Test to Fully Exercise All Control Rod Banks Multiple Times Following Outages and Prior to Reactor Startups**
- **New Test Implemented Following Outage 2R25 in Fall 2025.**



Dropped Control Rod Event Corrective Actions (continued)

Conclusion: The corrective actions completed by DCCP in response to two Control Rod drop events were appropriate, and suitable long-term corrective actions were initiated.

Recommendations:

- The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and is recommended for closure: *The DCISC should follow up on future corrective actions that may be initiated in response to an August 2024 dropped rod event. (D11/24FF-3.13)*
- The DCISC should review current practices and recent results for the surveillance testing of Control Rod Drive Mechanisms during a future Fact-Finding Meeting. *(D7/6/26FF-3.1)*



Major Project Planning

- **Major Projects Planned for 1R26 and 2R26 Include (Both Units):**
 - **Condenser Pressure Transmitter Relocation**
 - **Main Generator Relay Replacement (with digital relays)**
 - **Tri-Conics Control System Migration (new hardware and software)**
 - **Condenser Water Box Coatings**
 - **Main Annunciator System Replacement (with digital system)**
 - **Circulating Water Pump Motor rewind**
 - **Fuel Transfer Cart Upgrades**
 - **Feedwater Heater Replacement Preparatory Activities (minor electrical conduit relocations)**



Major Project Planning (continued)

- **Major Projects Planned for 1R26 (Unit 1 Only):**
 - **230kV Relay Replacement with Digital Relays**
 - **Intake Screen Wash Piping Replacement**
 - **Condenser Saltwater Expansion Joint Replacements**
 - **Reactor Vessel and Baffle Bolt In-Service-Inspection**
 - **500kV Tie Line Relay Replacement with Digital Relays**
 - **Auxiliary Transformer 1-1 Radiator Replacement**
- **Major Projects Planned for 2R26 (Unit 2 Only):**
 - **500kV Tower 5/4 replacement**
 - **Advanced Reactor Fuel Features**
 - **500kV Lightning Arrestor replacement**



Major Project Planning Update (continued)

- Major Projects Planned for 1R27 (Spring 2028) and 2R27 (Fall 2028) Include:
 - Feedwater Heater (FWH) Replacements (pending extended ops authorization)
 - Circulating Water Pump Mechanical Seal Replacements
 - Condensate Booster Pump Baseplate Replacement (Unit 1 only)

Conclusion: The DCISC found that PG&E's planning for upcoming major plant projects was being effectively managed and projects important to safety were being pursued for implementation.

Recommendation: The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and remains open: *The DCISC should continue to follow and support DCCP's plans for future Feedwater Heater replacements. (D4/26FF-3.10A)*



Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring

- **DCPP Submitted the Following Annual Reports to the NRC in April 2026:**
 - **2025 Annual Radioactive Effluent Release Report (ARERR)**
 - The ARERR describes the measured/calculated quantities of radioactive gaseous effluents, liquid effluents, and direct radiation released from the plant.
 - **2025 Annual Radiological Environmental Operating Report (AREOR)**
 - AREOR measures and assesses the levels of radiation or radioactivity in the environment related to operation of DCPP and verifies that DCPP operated within its design parameters.



Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring (continued)

- **2025 Annual Radioactive Effluent Release Report (ARERR)**
 - The ARERR measured and described the gaseous and liquid radiological effluent releases from DCPD Units 1 and 2, direct radiation dose to a member of the public, and shipments of solid radioactive waste during 2025.
 - The results showed that the doses associated with plant effluent releases were much less than the DCPD Technical Specification and NRC limits.



Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring (continued)

- **2025 Annual Radioactive Effluent Release Report (ARERR) Effluent Summary:**

<u>Effluent Type</u>	<u>Radiation Dose</u>	<u>Limit</u>	<u>Percent of Limit</u>
Liquid (total body)	0.000385 millirem	3 millirem	0.013%
Gaseous (gamma)	0.00265 millirem	10 millirad	0.027%
Gaseous (beta)	0.00100 millirem	20 millirad	0.005%



Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring (continued)

- **2025 Annual Radiological Environmental Operating Report (AREOR)**
 - The AREOR described the results of the Radiological Environmental Monitoring Program (REMP), which measures and assesses the levels of radiation or radioactivity in the environment related to operation of DCPD and verifies that DCPD operated within its design parameters during 2025.
 - The results showed that ambient direct radiation levels in the DCPD offsite environs did not change in 2025 and were within the pre-operational background range.
 - The annual offsite radiological dose received by any member of the general public from plant operations was less than one millirem



Reports on 2025 Radiation Releases and Radioactive Environmental Monitoring (continued)

Conclusion: The DCPD Radioactive Effluent Release Program and the Radiological Environmental Monitoring Program appeared satisfactory in calculating, monitoring, and measuring the impacts of radioactivity from DCPD on the environment. During 2025, there were no abnormal releases of radioactivity, and data contained in the reports demonstrated that DCPD continued to have no detectable radiological impact to the environment surrounding the plant.



Observe Leadership Alignment Meeting

- Meeting of All Supervisors and Above to Receive Presentations from Station Senior Leaders
- Observed Good Participation in “Teaching and Learning Activity”
- Meeting Well Facilitated and Discussions Appeared Open and Effective

Conclusion: A July 6, 2026, Leadership Alignment Team meeting with all station supervisors and leaders was effectively conducted and allowed opportunities for significant discussion of important topics among all attendees.



Condensate and Feedwater System Health

- **System Collects Steam from Turbines to Condense to Water and Moves Water to Steam Generators**
- **Consists of Condensers, Condensate and Feedwater Pumps, Feedwater Heaters**
- **System Important to Safety, but Only Portions Classified as Safety-Related**
- **System Health Reports Reviewed, in Good Health Overall**
- **Issues Included:**
 - **Feedwater Heater Aging (reviewed separately by the DCISC)**
 - **Minor Condenser Tube Leaks**
 - **Minor Pump Issues Being Addressed**



Condensate and Feedwater System Health (continued)

- Discussed the Effects of Climate Change on Condenser and in General
 - Evaluation for CA Coastal Commission (2024) Reviewed in January 2025
 - DCPD Also Completed a Climate Vulnerability Assessment in 2025

Conclusion: DCPD's Condensate and Feedwater Systems on both units were healthy with no major issues affecting operations (other than aging Feedwater Heaters).

Recommendation:

The DCISC should review the results of DCPD's 2025 Climate Vulnerability Assessment and the associated Climate Hazard Information and Projections Report during a future Fact-Finding Meeting. (D7/6/26FF-3.5)



Station Update

- **Station Update:**
 - **Status of Institute of Nuclear Power Operations Indicators**
 - **Status of Cause Evaluation for Startup Transformer 2-1 Bus Duct Failure**
 - **Refueling Outage 1R26 Preparations**

Conclusion: DCPD's update on recent station activities was informative, and no new major issues were identified.



Reactor Vessel Sample Capsule B Report

- **PG&E and Westinghouse Briefing on Capsule B Report, Issued on June 29**
- **Attendees Included Dr. Mark Kirk and Department of Water Resources Staff**
- **Report Confirmed that Unit 1 Continues to Meet NRC Requirements for Reactor Vessel Integrity and Remains Safe for at Least 60 years of Operation**
- **Following this Meeting, a Series of Follow-up Meetings and Information Exchanges were Conducted to Support Evaluations by Members and Dr. Kirk**



Reactor Vessel Sample Capsule B Report (continued)

Conclusion: DCPD presented the results of testing and evaluation of material specimens contained in Reactor Vessel Embrittlement Sample Capsule B, which was removed from the vessel on April 21, 2025. PG&E staff concluded that the results confirmed that Unit 1 continues to meet NRC requirements for Reactor Vessel integrity and remains safe for at least 60 years of operation. The DCISC will continue to perform its own reviews of the test results which will be further discussed at its October 2026 Public Meeting.

Recommendation:

The following previous recommendation to PG&E was reviewed in this Fact-Finding Meeting and remains open: *Recognizing that Capsule B test results are not necessary to assure Unit 1 Reactor Pressure Vessel embrittlement acceptability, the DCISC recommends that the Capsule B be removed at the next outage in 2025 and that a re-evaluation of reactor vessel integrity, including both PTS and USE, be performed once the new data from Capsule B become available.* (AR23/24AR23-2)



Meet with DCPD Officer

- Meeting with Paula Gerfen, Senior Vice President Business and Chief Nuclear Officer
- Discussed the Fact-Finding Agenda Items and Other Items of Mutual Interest

Conclusion: The regular meetings between DCISC and DCPD Officers and Directors continued to be beneficial for both organizations.



Fire Protection Program and System Health

- **Program Manages the Prevention and Response to Fires at the Station**
- **System Provides Fire Detection, Alarms, and Suppression**
- **Program Health Report Reported “Green” for Most Major Areas**
- **The Area of Equipment Performance Was “White” Due to a High Backlog of Maintenance Work Orders Older than 24 Months**
- **The Team Requested a List of Open Work Orders (23) Classified as “Adverse to Quality” and Reviewed for Significance**
- **Only a Small Number Represented Actual Equipment Problems**



Fire Protection Program and System Health (continued)

- **Low Numbers of Fire Watches and Fire Door Issues**
- **No Major Issues with Fire Water Supply System**

Conclusion: DCPD's Fire Protection Program and Fire Protection Systems were in excellent health overall. Minor equipment issues were being appropriately tracked for resolution.



Safety-Related Fuse Failure

- **Follow-Up Item from February 2026 Public Meeting**
- **Fuse Failed in Eagle 21 Protection System in 2025, Investigations Found Fuse Had Not Been Replaced Since 2009**
- **NRC Green Non-Cited Violation for Failure to Implement Preventive Maintenance**
- **Issue Limited to Small Number of Fuses for Which a Preventive Maintenance Change Request Was Not Properly Implemented for Replacing the Fuses Every 10 Years**
- **Subject Fuses Replaced and Maintenance Strategy Implemented**

Conclusion: DCPP appropriately investigated and responded to a July 2025 event in which a safety-related fuse failed in the Eagle 21 Protection System.



Observe Nuclear Safety Oversight Committee Meeting

- **Remotely Observed the Nuclear Safety Oversight Committee' s (NSOC' s) Exit Meeting on August 13, 2026**
- **External Committee of Five Outside Executive-Level Industry Peers**
- **Advises the Chief Nuclear Officer on Safety Policy and Provides an Independent Perspective on Plant Performance**
- **Typically Meets and Visits DCCP Three Times per Year**

Conclusion: The DCCP Nuclear Safety Oversight Committee appeared to be thorough and comprehensive in their evaluations and candid in their reports.



Per Peterson, Member, with Richard McWhorter, Consultant

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- 11. Observe Nuclear Safety Oversight Committee Meeting (Remote)**



Najmedin Meshkati, Member, with Lisa Darden and Ferman Wardell, Consultants

- 1. Reactor Vessel Embrittlement Sample Capsule B Report**
- 2. Meet with Nuclear Regulatory Commission Resident Inspector**
- 3. Meet with DCPD Officer**
- 4. Walkthrough of DCPD Dynamic Procedures**
- 5. Boric Acid Corrosion Control Update**
- 6. Emergency Plan Drill Observation Pre-Drill Brief**
- 7. Station Update, Including:**
 - Discuss Documents for DCPD Decision Making Concerning Plant Operations**
 - Unit 2 12kV Bus Duct/Start-Up Transformer Cause Evaluation Update**
- 8. PG&E Employee Corporate Safety Culture Survey Results Review**
- 9. Emergency Plan Drill Observation**



Reactor Vessel Sample Capsule B Report

- PG&E and Westinghouse Briefing on Capsule B Report, Issued on June 29
- Additional Attendees Included California Energy Commission Chairman David Hochschild, and Staff Members Justin Cochran, Robert Chun, and Drew Bohan
- Report Confirmed that Unit 1 Continues to Meet NRC Requirements for Reactor Vessel Integrity and Remains Safe for at Least 60 years of Operation
- Following this Meeting, a Series of Follow-up Meetings and Information Exchanges were Conducted to Support Evaluations by Members and Dr. Mark Kirk



Reactor Vessel Sample Capsule B Report (continued)

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Meet with Nuclear Regulatory Commission Resident Inspectors

- Meeting with Mahdi Hayes, NRC Senior Resident Inspector, and Eli Garcia, NRC Resident Inspector
- Discussed various topics including:
 - DCPD Unit 2 Start-Up Transformer Bus Duct Arc Event and Cause Evaluation Progress
 - Reportability Requirements in various Code of Federal Regulations and NRC Guidelines
 - The DCPD Main Annunciator System Replacement Project
 - The NRC Review of the DCPD Reactor Vessel Embrittlement Sample Capsule B Analysis Results

Conclusion: The meeting with the NRC Resident Inspectors was beneficial, and the DCISC should continue these meetings.



Meet with DCPD Officer

- Meeting with Paula Gerfen, Senior Vice President Business and Chief Nuclear Officer
- Discussed the Fact-Finding Agenda Items and Other Items of Mutual Interest

Conclusion: The regular meetings between DCISC and DCPD Officers and Directors continued to be beneficial for both organizations.



Walkthrough of DCPD Dynamic Procedures

- DCPD has developed, tested, and prototyped the implementation of Dynamic Procedures (DP) for Operations equipment rounds.
- DPs ensure that conditions are met prior to advancing to the next step.
- DPs ensure automatic entry of recorded measurement data into the station's Data Historian, Notifications are created for off normal conditions.
- Operators can take pictures and submit Notifications of issues they identify while still in the field.
- Apple iPads are the preferred tablet technology since iPads do not require fan cooling, eliminating internal contamination from travel in radiation areas.
- NextAxiom software is used and supported by a robust Users Group.



Walkthrough of DCPD Dynamic Procedures (continued)

- DPs show potential to reduce human errors and improve Operator performance.
- Industry studies have concluded that computer-based procedures (CBPs) can support enhanced Operator performance (finish tasks faster, carry a lower cognitive workload, and make fewer transition errors) if implemented well.
- Poorly implemented CBPs can reduce human reliability.
- The NRC now maintains review guidelines for CBP systems in Part II of NUREG-0700, Revision 3 (2020).
- Human factors issues and lessons learned, NRC reviews, and industry group involvement are important areas of interest to the DCISC.



Walkthrough of DCPD Dynamic Procedures (continued)

Conclusion: The Dynamic Procedures in use and under development at DCPD show promise for reducing human error and improving Operator performance in Operator rounds and other evolutions. Realizing that promise depends on a systematic, independent human factors evaluation of the system's design, implementation, and failure modes, not on Operator acceptance alone.



Walkthrough of DCPD Dynamic Procedures (continued)

Recommendation: Human factors issues and lessons learned, NRC reviews, and industry group involvement associated with Dynamic Procedures should be included in the DCISC Open Item List (OIL) for review at future fact-finding meetings. Relevant industry groups include EPRI, INPO, and vendor users' groups. At a minimum, the review should address the following:

- (a) Regulatory review. Has the NRC reviewed or inspected DCPD's use of Dynamic Procedures? To what extent has PG&E applied the CBP system guidelines in NUREG-0700, Rev. 3, and the CBP review guidance in NUREG/CR-6634 to their design and implementation?
- (b) Human factors and human-computer interface assessment. How are human factors and human-computer interface issues for Dynamic Procedures assessed, by whom, and on what timeline? Is the assessment performed solely by the Operators who use the system, or does it include independent human factors verification and validation? Specific topics include:
 - situation awareness
 - the allocation of functions between the software and the Operator
 - crew communication
 - management of multiple concurrent procedures
 - backup procedures and the Operator's transition to them if the system fails
- (c) Industry participation and operating experience. Besides the vendor's users' group, does PG&E participate in EPRI working groups relevant to Dynamic Procedures, such as the Work Planning Users Group (WPUG)? Does DCPD follow Procedure Professionals Association (PPA) guidelines (e.g., PPA AP-907-005), and do its procedure writers hold PPA certification or membership? Is operating experience on implementing Dynamic Procedures available through INPO or WANO, and how has DCPD used it? D7/28/26FF-3.4



Boric Acid Corrosion Control (BACC) Update

- DCPD currently has a high number of BACC leaks classified as “LK3” – Dry, Discolored or Excessive Leaks. The associated Plant Indicator is “Red” (Unhealthy).
- Identified by the DCISC at the January 2026 fact-finding meeting where the following recommendation was opened:
 - PG&E should devote additional resources as needed to the BACC Program and to repair the high number of leaks classified as “LK3” in the plant with the aim of achieving the Boric Acid Leak Maintenance performance indicator goals.
- The DCPD Boric Acid Leak Maintenance performance indicator goal came from the Institute of Nuclear Power (INPO) and is in line with industry guidance.
- The actual number of LK3 leaks was about twice the goal for both units in January 2026 and continued at this level through July with a slight decreasing trend on Unit 1 and a slight increasing trend on Unit 2.



Boric Acid Corrosion Control (BACC) Update (continued)

- DCPD plans to repair approximately one third of the Unit 1 LK3 leaks during the upcoming fall Unit 1 Refueling outage 1R26. This will not decrease the number of Unit 1 LK3 leaks to less than or equal to the established goal.
- DCPD does not remove LK3 leaks from the list after maintenance is performed until a visual verification can be performed after placing the component back in service. Some of these components are in radiological restricted areas. This is generally different than the industry.
- DCPD had not established an action plan to address the “Red” performance indicator and drive improvement to “Green” (Healthy).



Boric Acid Corrosion Control (BACC) Update (continued)

Conclusion: DCCP's Boric Acid Corrosion Control Program was being implemented in accordance with applicable industry guidelines; however, the number of leaks being tracked by the program continues to be unusually high and there is no formal action plan to improve performance. The DCISC recommendation for PG&E to devote additional resources as needed to the Boric Acid Corrosion Control Program and to repair the high number of leaks classified as "LK3" (inactive leaks) in the plant should remain open. PG&E should also develop a proactive action plan to reduce the high number of leaks classified as "LK3" in the plant with the aim of achieving the Boric Acid Leak Maintenance performance indicator goals.



Boric Acid Corrosion Control (BACC) Update (continued)

Recommendations:

The DCISC should meet with the members of the Boric Acid Review Team (BART) at a future fact-finding meeting to review the overall performance of the Boric Acid Corrosion Control Program as well as the status of the Red indicator associated with LK3 leaks. (D7/28/26FF-3.5)

The following previous recommendation to PG&E was reviewed in this Fact-Finding Meeting and remains open: *PG&E should devote additional resources as needed to the Boric Acid Corrosion Control Program and to repair the high number of leaks classified as “LK3” in the plant with the aim of achieving the Boric Acid Leak Maintenance performance indicator goals. (P1/26FF-3.3)*

PG&E should develop a proactive action plan to reduce the high number of leaks classified as “LK3” in the plant with the aim of achieving the Boric Acid Leak Maintenance performance indicator goals. (P7/28/26FF-3.5)



Emergency Plan Drill Observation Pre-Drill Brief

- DCPD provided a pre-evolution brief to the fact-finding team members for the Emergency Plan Drill the next day.
- The fact-finding team learned the expectations for observers when at the different PG&E facilities, the pre-drill conditions, and was able to ask questions to support a successful observation.

Conclusion: PG&E's pre-drill brief to the DCISC Fact-Finding Team provided a helpful review of expectations and information prior to the execution of the Emergency Plan drill.



Station Update

- Discussed three DCISC Open Item List (OIL) topics:
 1. From the October 2025 Public Meeting: *The Station Manager reported that DCPD has clear guidance with other entities within PG&E that addresses the interface and the decision-making associated with the plant and the down time of the plant and no authorization or clearance from outside DCPD is required. DCPD offered to provide Dr. Meshkati with a copy of the guidance document at a future fact-finding meeting.* (10/25PM#20)
 2. The DCISC should follow-up on the Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure that occurred on May 4, 2026. (D5/26FF-3.3A)
 3. The DCISC should follow-up on the rationale for performing a Cause Evaluation vs. a Root Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure that occurred on May 4, 2026. (D5/26FF-3.3B)



Station Update (continued)

1. Guidance documents for DCPD to reduce power, realign the plant, or shutdown:

- The Station Manager provided the fact-finding team three specific DCPD procedures designated for station decision-making to ensure the safety of the station from internal and external events (e.g., wildfires).
- Emphasis was provided in the area where the Shift Manager and Operations team make these decisions and examples were provided.
- This provided the fact-finding team the confidence that DCPD has sufficient guidance and autonomy to make the operational decisions necessary to protect the plant.
- Critiques are held after events where these procedures are used with PG&E Transmission and DCPD personnel to identify procedure issues or improvement opportunities.



Station Update (continued)

2. Status of the Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure:

- The cause evaluation team continued to investigate the failure and proposed replacement of the feeder cable stress cones based on a potential anomaly (insulation sponginess).
- Stress cones are specialized geometric devices used in medium- and high-voltage cable terminations to prevent insulation failure by controlling the electric field.
- Installation is infrequently performed and requires precision, so DCPD set up a practice shop to ensure proficiency. Supported by Training and Engineering.
- Transformer 2-2 was removed from service the week of July 20, 2026, and the stress cones were successfully replaced.
- As of this fact-finding meeting, the cause evaluation team was evaluating the removed stress cones and including them in their assessment.

(continued) 17



Station Update (continued)

3. Rationale for performing a Cause Evaluation vs. a Root Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure:

- The details and sequence of events of the failure were reviewed with the fact-finding team.
- Pertinent to the discussion was a review of the security video of the bus duct and transformer with the low-level arc/flame that was observed.
- The video was the evidence DCPD used to determine the event did not raise to the level of an industry High Energy Arc Fault (HEAF) event such as presented in NRC Information Notice 2017-04, “*High Energy Arcing Faults in Electrical Equipment Containing Aluminum Components.*”
- DCPD committed to performing a thorough cause evaluation. The team’s identification of a potential contributing issue with the feeder cable stress cones supports this.



Station Update (continued)

Conclusion: DCCP's update on recent station activities was informative, and no new major issues were identified. The Fact-Finding Team appreciated the Station Manager providing information to address some of the DCISC Open Item List items.



Station Update (continued)

Recommendations:

The following DCISC Open Action Item from the October 2025 Public Meeting was reviewed in this Fact-Finding Meeting and should be closed: *He [Mr. Rogers] reported DCPD has clear guidance with other entities within PG&E that address the interface and the decision-making associated with the plant and the down time of the plant and no authorization or clearance from outside DCPD is required. He offered to provide Dr. Meshkati with a copy of the guidance document at a future fact finding.* (10/25PM#20)

The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and remains open: *The DCISC should follow-up on the Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure that occurred on May 4, 2026.* (D5/26FF-3.3A)

The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and should be closed: *The DCISC should follow-up on the rationale for performing a Cause Evaluation vs. a Root Cause Evaluation for the Unit 2 Start-Up Transformer 2-2 bus bar failure that occurred on May 4, 2026.* (D5/26FF-3.3B)



PG&E Employee Corporate Safety Culture Survey Results Review

- DCISC was informed of the corporate-wide survey of employee safety culture that was completed in 2025 during the January 2026 fact-finding meeting. This was a follow-up to review the results.
 - The survey is related to a California requirement for utilities.
 - Used by 1400 companies and targeted to measure employee safety perception.
 - PG&E overall score was 84 which is considered high (positive – scored higher than 84% of the companies surveyed).
 - 22,430 employees completed the survey out of 28,400 total employees.
- The fact-finding team posed questions to help clarify the intent of many survey questions as well as the results, and of particular interest, obtaining data specific to DCP.



PG&E Employee Corporate Safety Culture Survey Results Review (continued)

- As DCISC's mandate concerns the safety of operations at DCP, requests were made to obtain DCP specific data from the survey:
 - The survey separated results for each business group.
 - The "Generation" group had 782 respondents, with Nuclear Generation accounting for 557 of those (71%).
 - "Generation" was a high performing group in general, but several components scored notably lower.
 - Will allow DCISC to assess employee safety perceptions at DCP directly and determine whether the strong Generation-wide results hold for the nuclear workforce.
 - Will also help identify any areas warranting further attention as DCP operates beyond its original licensing period.



PG&E Employee Corporate Safety Culture Survey Results Review (continued)

Conclusion: The results of a PG&E corporate-wide survey of employee safety culture completed in 2025 were generally positive. The DCISC continues to work with DCPD to better understand the specific results that are reflective of DCPD culture.

Recommendation: The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and remains open: *The DCISC should review the results of a PG&E employee safety culture survey following its completion.* (D1/25FF-3.9A)



Emergency Plan Drill Observation

- Observed key phases of the July 29, 2026 DCPD emergency exercise drill.
- The drill was a Hostile Action-Based (HAB) Exercise – a specialized emergency preparedness exercise that tests how a nuclear power plant, local law enforcement, and off-site agencies coordinate their response to a simulated security threat or armed attack.
- The drill resulted in a simulated radioactive release to the environment.
- The purpose of the drill was to develop and maintain the Emergency Response Organization's (ERO's) key skills to protect public health and safety in the unlikely event of a radiological emergency.
- Since this was a drill, coaching by Drill Controllers and Emergency Planning staff was permitted.
- The Exercise was designed to evaluate the proficiency of DCPD personnel in implementing the principal Emergency Plan functions in response to a radiological emergency.



Emergency Plan Drill Observation (continued)

- The plant simulator acted as the Control Room with the simulator computer controlling all indications at the Emergency Response Facilities (ERFs).
- Participants included the NRC, Sheriff's Watch Commander, County Emergency Operations Center, and the State of California Office of Emergency Services Warning Center participated.
- Objectives evaluated include:
 - Communications between onsite and offsite ERFs
 - Coordination with offsite response organizations
 - Dissemination of information to the public via media channels
 - Operational and engineering assessment of accident sequences
 - Engineering assessment, repair plan development, and repair of critical equipment
 - Mitigative action implementation through the simulated repair of equipment
 - Protection of workers (radiological or physical) during emergency response

(continued) 25



Emergency Plan Drill Observation (continued)

- Exercise scenario summary:
 - Initial conditions: Both units 100%; plant holiday with staffing reduced to essential personnel
 - Gunboat attack on Protected Area (PA)
 - Site Area Emergency (SAE) declared
 - Helicopter incursion on PA resulting in a fire
 - Explosion in the PA
 - An explosive devise is found in the PA; Offsite Law Enforcement assistance provided
 - Unit 1 Emergency Bus 'H' trips
 - Explosion in Unit 1 Containment causing General Emergency Conditions:
 - Large Break Loss of Coolant Accident (LOCA)
 - 3% Fuel damage
 - Loss of Containment Integrity
 - General Emergency Declared and Protective Action Recommendations (PARs) made

(continued) 26



Emergency Plan Drill Observation (continued)

- Observed Simulator Control Room, Emergency Operations Facility (EOF) on Kansas Avenue, and the Joint Information Center (JIC).
- JIC mock news conferences were observed with a PG&E Spokesperson, a PG&E Public Information Officer, an SLO County Spokesperson, a SLO County Public Health Physicist, and mock media personnel. Numerous mock press releases were made by PG&E, SLO County, and the City of Pismo Beach. All appeared accurate and publicly understandable.
- Observed the EOF post-drill critique upon conclusion of the exercise drill. The critique was sufficiently critical and inclusive of all attendees.
- Overall, exercise objectives were met.



Emergency Plan Drill Observation (continued)

Conclusion: The July 29, 2026, DCPD Emergency Exercise Drill appeared well designed and implemented. The scenario involved a sufficiently large radioactive release to require declaration of a General Emergency. The critique observed was sufficiently critical and action items were documented. The DCISC should review the DCPD drill critique in a future fact-finding meeting when it is completed.

Recommendation: The DCISC should review the July 29, 2026 DCPD drill critique in a future fact-finding meeting when it is completed. (D7/28/26FF-3.9)



Najmedin Meshkati, Member, with Lisa Darden and Ferman Wardell, Consultants

1. Reactor Vessel Embrittlement Sample Capsule B Report
2. Meet with Nuclear Regulatory Commission Resident Inspector
3. Meet with DCPD Officer
4. Walkthrough of DCPD Dynamic Procedures
5. Boric Acid Corrosion Control Update
6. Emergency Plan Drill Observation Pre-Drill Brief
7. Station Update, Including:
 - Discuss Documents for DCPD Decision Making Concerning Plant Operations
 - Unit 2 12kV Bus Duct/Start-Up Transformer Cause Evaluation Update
8. PG&E Employee Corporate Safety Culture Survey Results Review
9. Emergency Plan Drill Observation

Diablo Canyon Power Plant

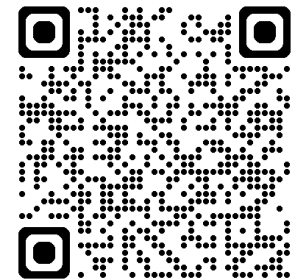
Unit 1 Capsule B Testing and Analysis Results



Together, Building
a Better California

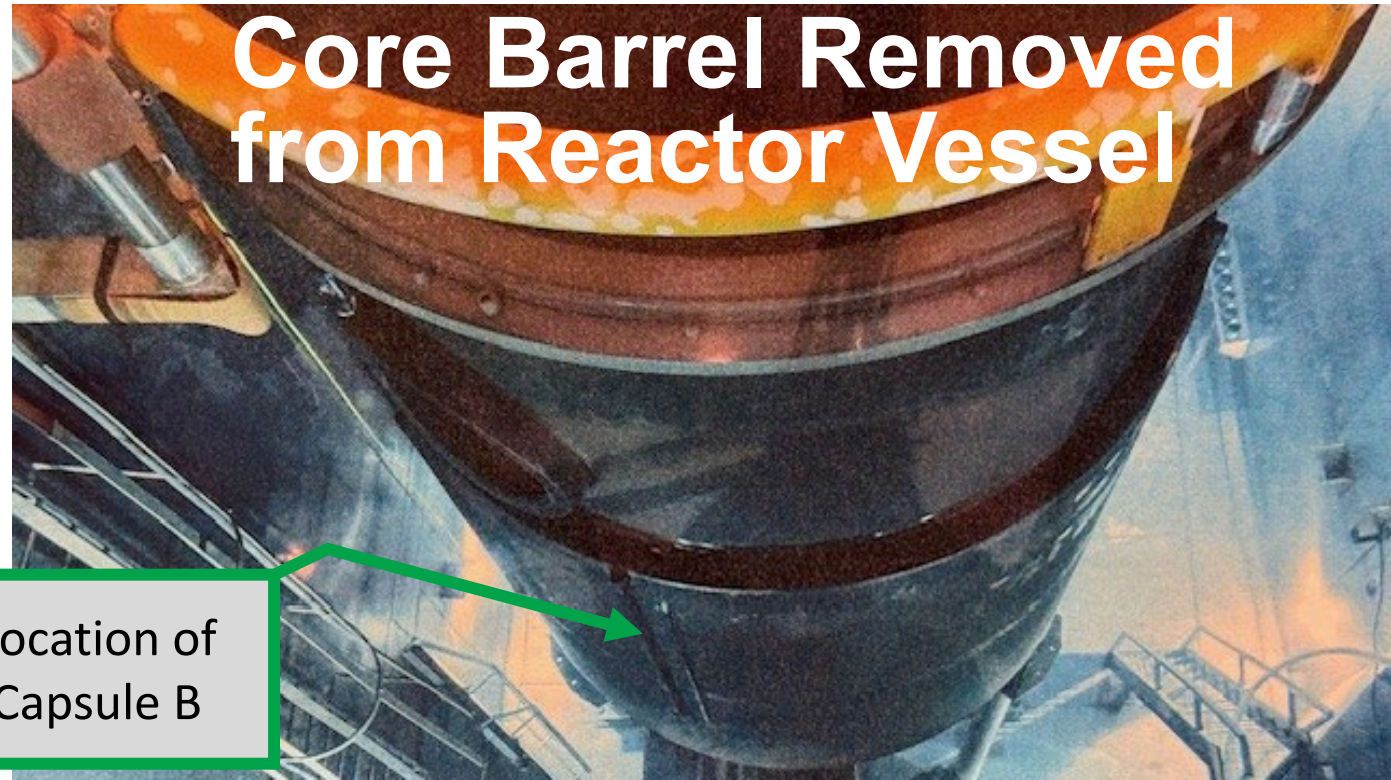
Summary

- Westinghouse performed extensive testing on Surveillance Capsule B per Nuclear Regulatory Commission (NRC) requirements and guidance
- Results further confirm Unit 1 continues to meet Federal reactor vessel integrity requirements and remains safe for at least 60 years of operation
- Results support NRC's previous June 2025 License Renewal Safety Evaluation Report conclusions and the existing reactor vessel integrity analysis of record
- Results submitted to NRC on June 29, 2026
- Independent NRC review confirmed results



Background: DCP Unit 1 Capsule B

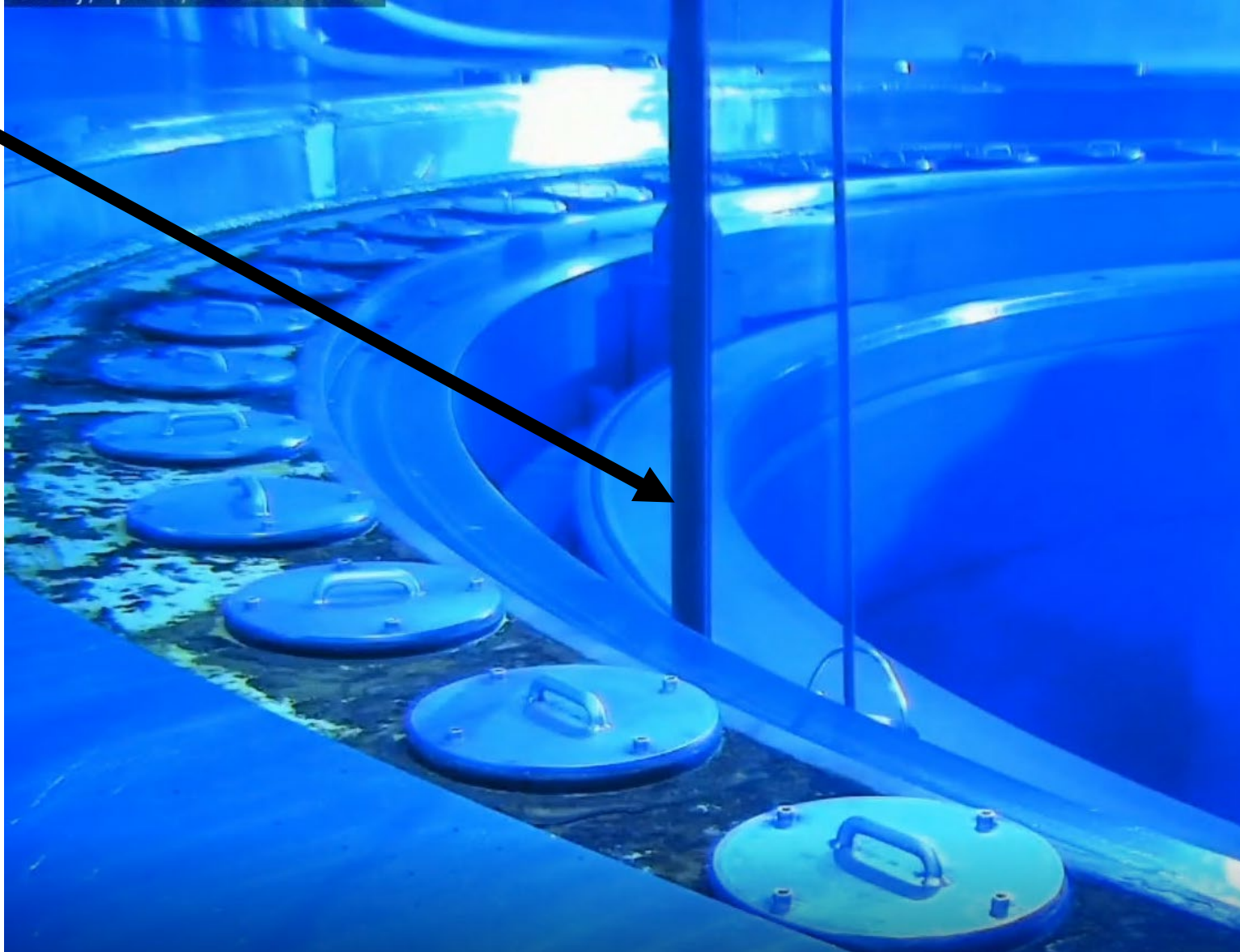
- PG&E complies with Federal regulations (10 CFR 50 Appendix H) that require a Reactor Vessel Material Surveillance Program for DCP Units 1 and 2
 - Periodically removes the surveillance capsules that contain metallic specimens to test and analyze; confirms the reactor vessels continue to remain safe
 - Existing capsule data evaluations also demonstrate both Units meet reactor vessel integrity criteria for at least 60 years through 2045
- To support 60 years of operation:
 - Unit 1 Capsule B data for confirmatory purpose
 - Already removed four Unit 2 capsules; no additional data needed to confirm acceptability
- **Capsule B successfully removed on April 21, 2025 to confirm previous predictions are still valid**



Location of
Capsule B

Background: DCP Unit 1 Capsule B

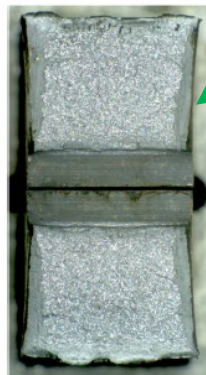
**Withdrawn
Capsule B**



DCPP Unit 1 Capsule B Testing and Analysis

- Capsule B contained weld and base metal specimens of the same heat # (batch) as the Unit 1 reactor vessel; also contained dosimetry and thermal monitors
- Rigorous testing conducted by Westinghouse per NRC-approved methods/standards
- Independent laboratory testing included over 50 specimens:
 - Charpy V-notch impact testing: 46 specimens
 - *Measures material toughness (how metal will perform under stress – bend/deform or shatter)*
 - *Testing uses a heavy swing hammer (pendulum) on the notched metal*

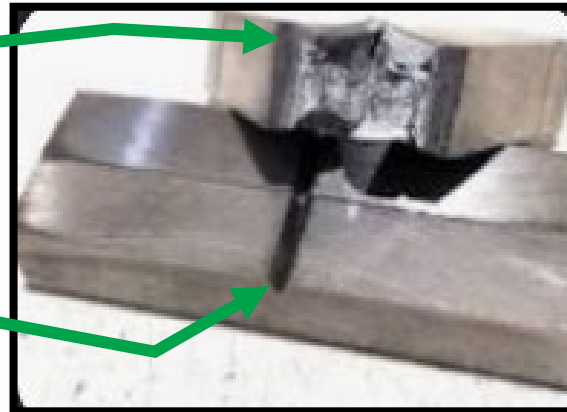
From Capsule B
Testing Report –
Unit 1 Vessel
Base Metal



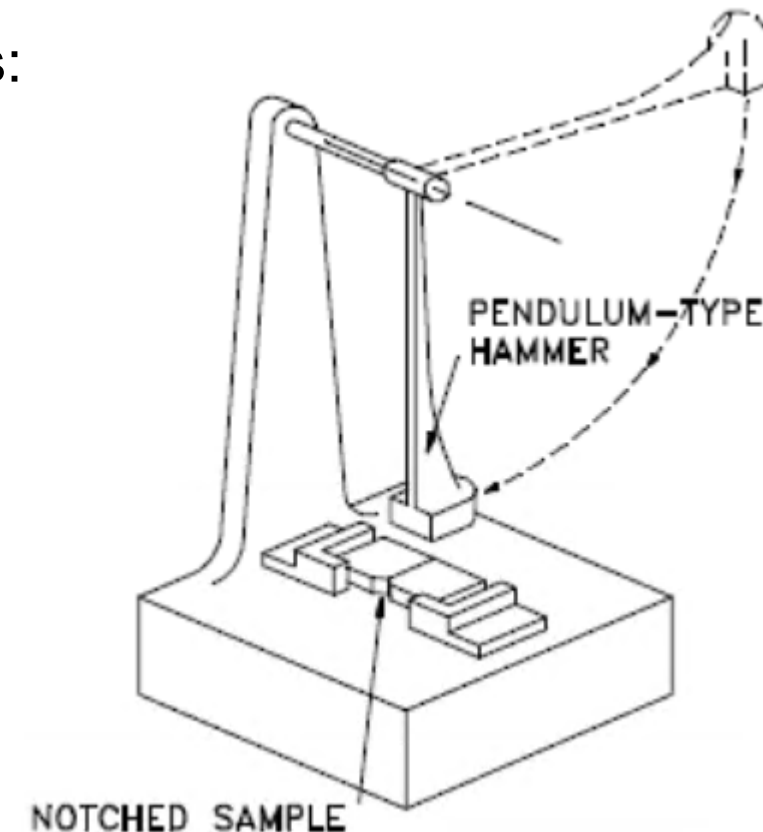
AT110, 180°F

Broken Halves

Notched "V"



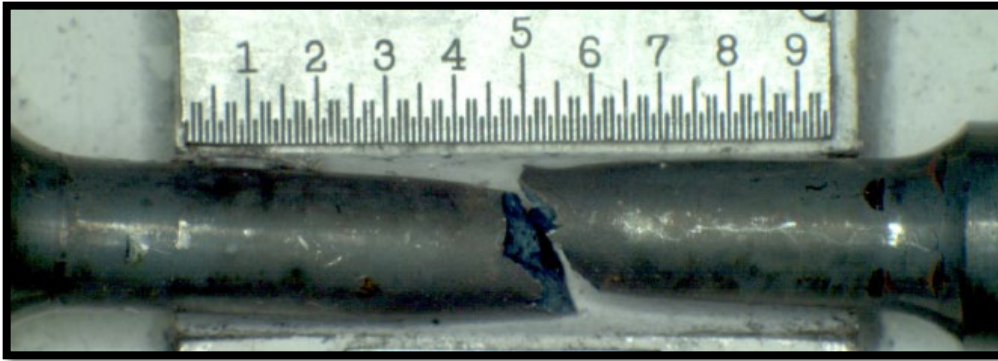
Illustrative Example



Example Test Set-up

DCPP Unit 1 Capsule B Testing and Analysis

- Independent laboratory testing included over 50 specimens (*continued*):
 - Tensile testing: 4 specimens
 - *Helps identify and quantify material characteristics used in analyses by understanding how the metal behaves when subjected to pulling forces*
 - *Tested by pulling each end of the metal*



**From Capsule B
Testing Report –
Unit 1 Vessel
Base Metal**

- Neutron dosimetry measurements
 - *Measures and calculates the absorbed dose of neutron radiation*
- Thermal monitor examination
 - *Informs the maximum temperature experienced by the test specimens during irradiation*

Testing Calibration and Quality Control

- Westinghouse used NRC-approved methods/standards
 - Ensures high-quality testing
 - *Highly calibrated equipment*
 - *Proven scientific steps*
 - Provides accurate and trustworthy data
- ✓ Equipment was calibrated using applicable standard and confirmed valid
- ✓ Independent checks confirmed measurement accuracy
- ✓ Impact and temperature equipment were regularly recalibrated
- ✓ Trained technicians verified test results
- ✓ Independent audits confirmed quality requirements were followed

Credibility Evaluation

- Credibility evaluations are performed to determine whether measurements are accurate, reliable, and scientifically sound enough to predict how the steel of the actual reactor vessel is aging due to radiation
- Non-credible data is still used, but additional safety margin is applied in predictions of material response
- Evaluated per the five criteria from Regulatory Guide (RG) 1.99, Rev 2: **no change to conclusions since previous analysis**
 - Credible for weld heat # 27204/flux Linde 1092 (used in the Intermediate & Lower Shell Longitudinal Welds 2/3-442A, B, C)
 - Includes use of Palisades “sister plant” data (same heat #/flux)

Neutron Fluence

- **Capsule B exposure equivalent:** $3.62 \text{ E}19 \text{ n/cm}^2$
 - ✓ Within 1-2 times (~1.7 times) the peak vessel fluence expected at 60 years (range $2.14 - 4.28 \text{ E}19 \text{ n/cm}^2$), per NUREG-1801, Rev. 2, XI.M31
- **Projection Methodology:** satisfies the requirements set forth in RG 1.190, Rev 0
 - ✓ Approved by the NRC; described in WCAP-18124-NP-A and WCAP-18124-NP-A, Rev 0, Supplement 1-NP-A
- **Limiting Fast Neutron Fluence Beltline Results at 60 years:** **same limiting locations as previous analysis**

Comparison to WCAP-18924-NP

- **WCAP-18924** is the reactor vessel integrity analysis of record
 - Cited in NRC's License Renewal Safety Evaluation Report
 - Previously evaluated by DCISC
 - Referred to as the “previous analysis” in these slides
- Previous analysis **conclusions remain valid** when considering Capsule B results – **DCPP Units 1 and 2 are safe to operate for at least 60 years:**
 - No change in credibility conclusions (i.e., results follow expected trends)
 - Neutron fluence projections roughly equate to those in previous analysis
 - RT_{PTS} inputs are bounding or equal to previous analysis (e.g., increase in the average Charpy V-notch 30 ft-lb temperature [ΔRT_{NDT}], margin term [M], unirradiated RT_{NDT} , etc.)
 - Upper Shelf Energy (USE) inputs are bounding or equal to previous analysis (e.g., copper content, 1/4T fluence at 60 years, unirradiated USE)



Unit 1 Vessel Integrity Independent Expert Reviews, 2012-2026

Independent technical expert conclusions and regulatory decisions have remained unchanged

● **Oct 2012**

DCISC 21st Annual Report (pg 327)

Concluded the two DCPD units can operate for 60 years, if renewed, without PTS posing a threat.

● **Sep 2015**

NRC request for additional information (RAI 4.2.2-4)

Verified Unit 1 satisfies the PTS screening criteria through 60 years of operation.

● **Jun 2024**

NRC embrittlement petition disposition (pg 2)

Found prior reviews addressed the issues raised and no significant new information warranted further action.

● **Oct 2024**

DCISC acceptance of Dr. Kirk's report (pg 392)

PG&E and DCISC technical experts concluded the Unit 1 vessel complies with NRC limits and is safe to operate.

● **Jun 2025**

NRC license renewal safety evaluation (Sec 4.2)

Concluded the Unit 1 vessel aging management, including USE and PTS, is acceptable for 60 years of operation.

● **Apr 2026**

NRC renewed Unit 1 license (Encl. 1 and 4)

Determined Unit 1 meets renewal safety standards. Renewed the license through Nov 2, 2044.

● **Sep 2026**

NRC assessment of Capsule B (Enclosure)

Confirmed the report satisfies regulatory requirements and previous conclusions remain unchanged.

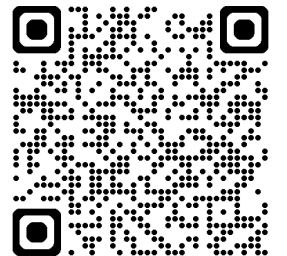
● **Oct 2026**

DCISC consultant Dr. Kirk's report (pg 3)

Confirmed reactor vessel integrity criteria will not be exceeded for at least 60 years of operation using NRC methods and supplemental analysis.

Key Takeaways

- Report (WCAP-19173-NP) satisfies NRC surveillance capsule withdrawal and testing requirements
- Comprehensive report shared with DCISC for review
- Surveillance capsule withdrawal and testing actions are satisfied for license renewal
- Ongoing dosimetry monitoring and periodic reactor vessel inspections continue through the 20-year renewed license period
- Both Units 1 and 2 are safe for at least 60 years of operations, as independently confirmed by NRC



Surveillance Capsule B from Diablo Canyon Unit 1

A Review of Data Reported in WCAP-19173-NP and its Implications

Mark Kirk

Phoenix Engineering Associates, Inc.

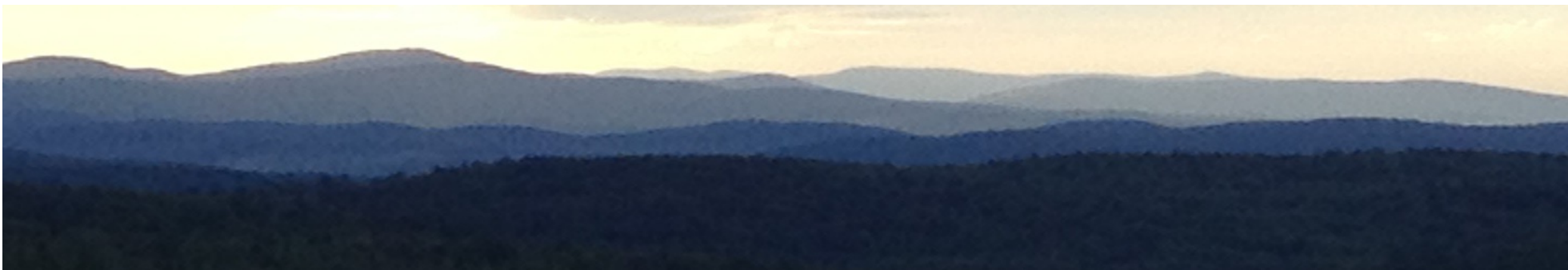
kirk@peaiconsulting.com

DCISC Meeting

Alvia Beach, CA
20 & 21 October 2026



**Phoenix
Engineering
Associates
Inc.**



Objectives

- Review data on weld wire heat 27204, including that provided by Capsule B testing, and evaluate analyses performed for PG&E by Westinghouse
 - April 2025: Capsule removed at end of Fuel Cycle 25
 - June 2026: Report sent to NRC (ADAMS ML26180A296)
- Assess impact of data from limiting weld wire heat 27204 on continued operability of Unit 1
 - Assessment relative to NRC's pressurized thermal shock (PTS) screening criteria (10 CFR 50.61)
 - Assessment relative to NRC's upper shelf energy (USE) screening criteria

Analysis of Capsule B from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program

Outline of Presentation

- ② Surveillance program summary
- ③ Data analysis methods
- ④ Assessment of data “credibility”
- ⑤ Assessment of Unit 1 relative to PTS screening criteria
- ⑥ Assessment of Unit 1 relative to USE screening criteria
- ⑦ Implications of possible future regulatory changes
- ⑧ Summary

Circled numbers are chapter titles in this report

Surveillance Capsule B from Diablo Canyon Unit 1 A Review of Data reported in WCAP-19173-NP and its Implications

Consultant's report to the Diablo Canyon Independent Safety Committee

Submitted by
Mark Kirk
Phoenix Engineering Associates Inc.
Unity, New Hampshire, USA

Date: 9 October 2026

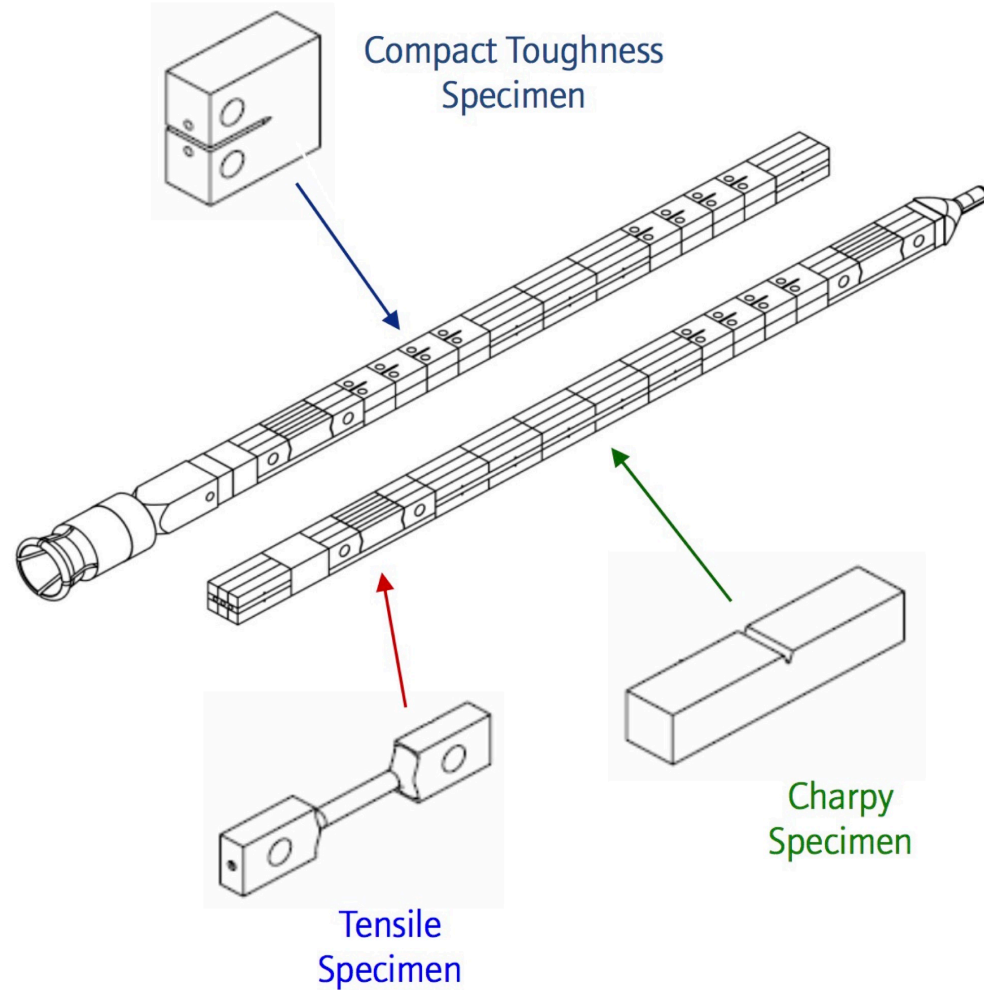


Phoenix
Engineering
Associates
Inc.

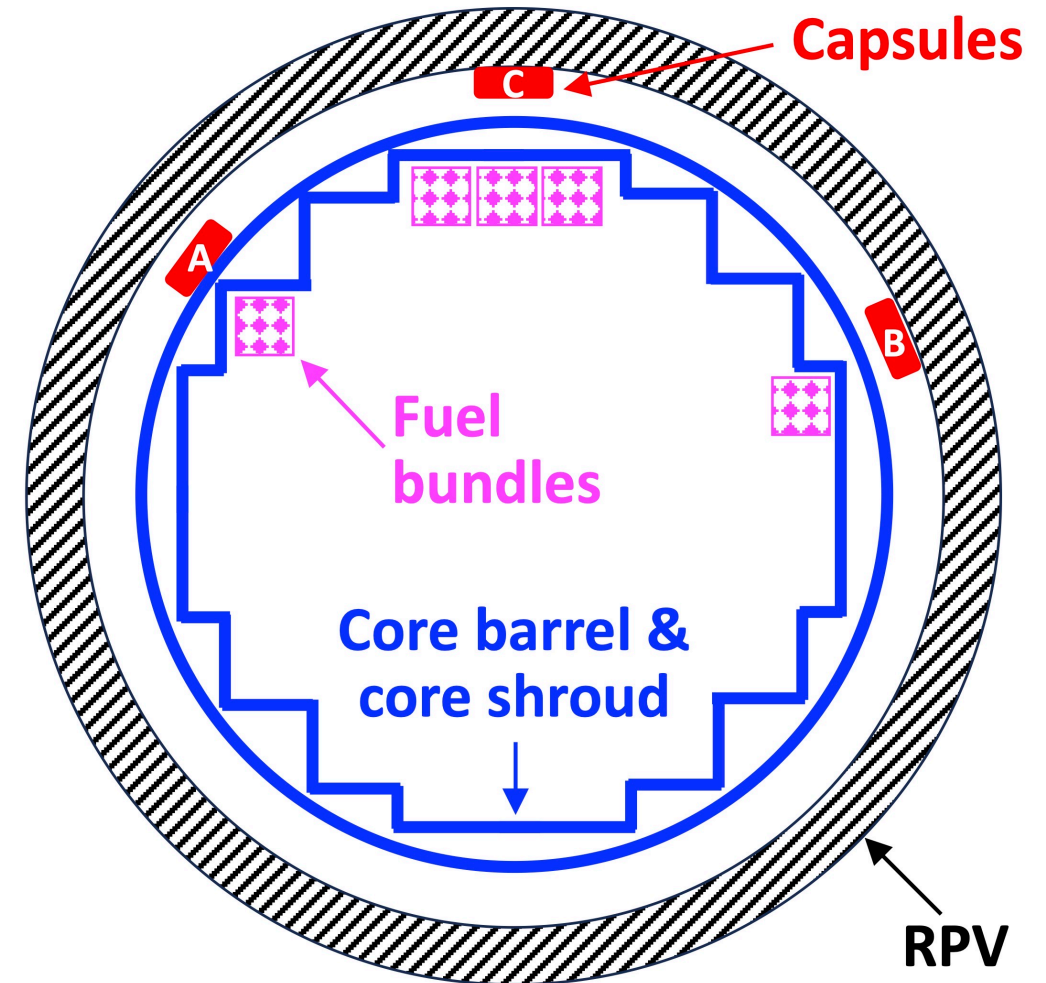
https://www.dcisc.org/download/events/51_14-f-Diablo-Canyon-Independent-Safety-Committee-20261020-Capsule-B-Review-of-Data-Reported-in-WCAP-19173-Its-Implications.pdf

② Surveillance Program

Surveillance Capsules Contain Specimens for Mechanical Property Testing



Capsules are Placed Closer to Core than is the RPV, so they Embrittle Sooner than RPV



② Original Surveillance Program

- Installed before reactor entered operation
- Capsules S, Y, and V were all tested by 2002
- This testing
 - Provided the data needed for operations through 40-years
 - Used all capsules containing the limiting weld (27204)

Capsule Designations	T, U, X, W, Z	S, Y, V
Lead Factors	3.57 for T & Z 1.17 for others	3.57 for S & Y 1.17 for V
Intermediate Shell Plate B4106-1	8 CVN, 1 Tensile, 2 WOL	---
Intermediate Shell Plate B4106-2	8 CVN, 1 Tensile, 2 WOL	---
Intermediate Shell Plate B4106-3	8 CVN, 1 Tensile, 2 WOL	8 CVN, 2 Tensile, 2 WOL
Weld Heat 27204, Linde 1092 flux	---	8 CVN, 2 Tensile, 2 WOL
Heat Affected Zone	---	8 CVN
Correlation Monitor Material	8 CVN	8 CVN

(Data from WCAP-8465)

② Supplemental Surveillance Program

- Inserted after fuel cycle 5
- Purpose
 - Supply data for operation > 40 years
 - Provide data for unforeseen contingencies
 - Provide data on the **limiting weld wire heat**
- Capsule B was recently tested

Material	Capsule A	Capsule B	Capsule C	Capsule D
Surrogate Weld from Nozzle Drop-Out. Weld heat (27204) and flux type (Linde 1092) same and flux lot (3714) as surveillance weld.	15 CVN, 3 Tensile	15 CVN, 3 Tensile	30 CVN, 3 Tensile	15 CVN, 3 Tensile
Lower Shell Plate B4107-1	15 CVN, 3 Tensile	15 CVN, 3 Tensile	15 CVN, 3 Tensile	15 CVN, 3 Tensile
Correlation Monitor Material	12 CVN	8 CVN	---	---
Surveillance Weld (from Capsule S), Heat 27204, Longitudinal Beltline Weld	---	2 WOL, 9 Charpy Inserts	---	8 Charpy Inserts
Weld 8B (Linde 80)	---	8 CVN	9 CVN	12 CVN
Weld 9B (Linde 0091)	---	7 CVN	9 CVN	11 CVN
Weld W7 (Linde 80)	---	7 CVN	8 CVN	11 CVN
Weld 72W (Linde 0124)	---	8 CVN	9 CVN	12 CVN

Note: Weld 8B, 9B, W7, and 72W are not used in the Unit 1 RPV. These welds were encapsulated to support a program conducted by the Electric Power Research Institute (EPRI).

(Data from WCAP-13440)

② Current Surveillance Program Status

- Capsule B

- ... provides data to a level of neutron exposure nearly twice that anticipated in the reactor after 60-years of operation
- ... fulfills all current NRC requirements through the end of the current 60-year license

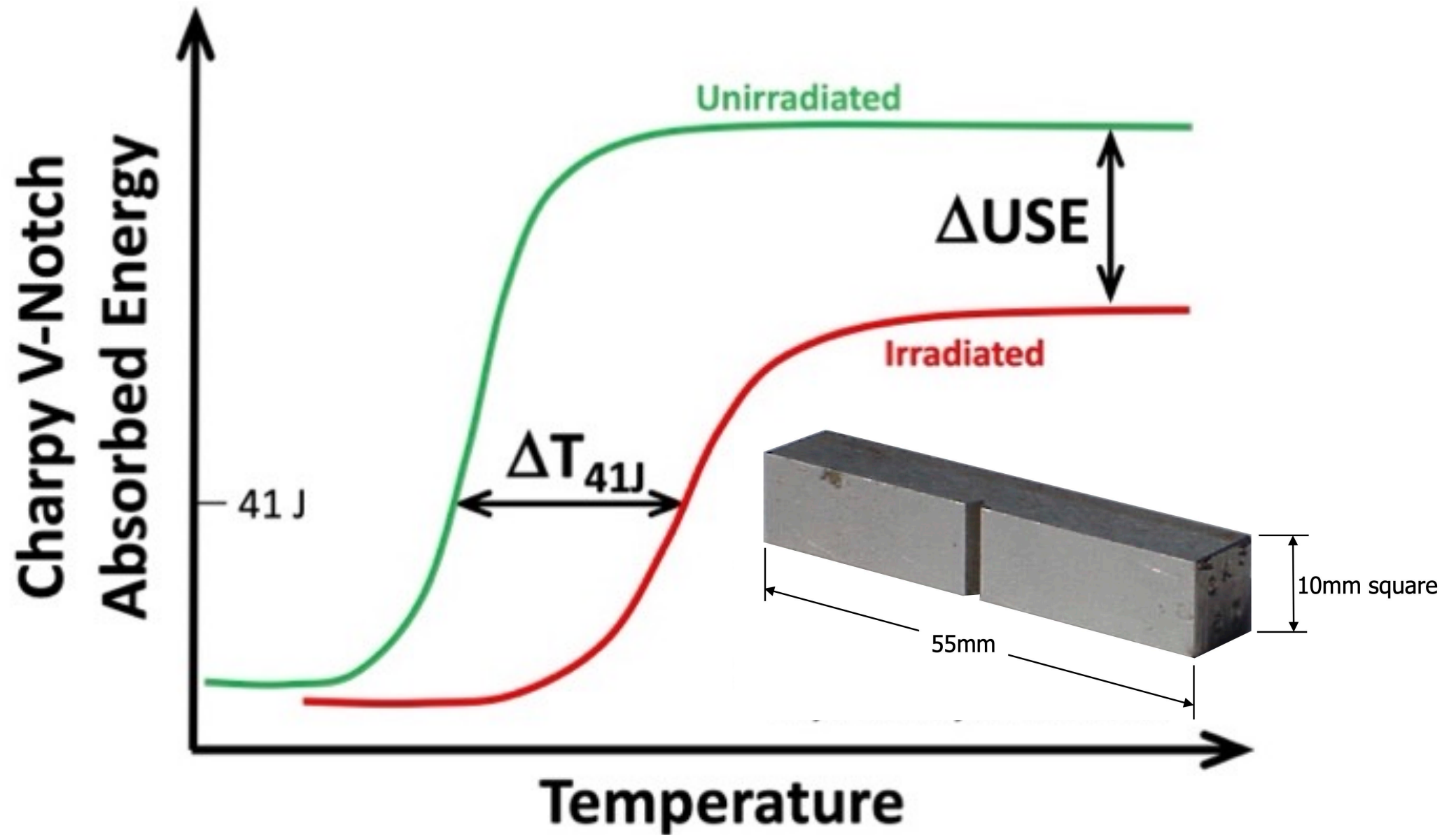
Status	Capsule	Insert Fuel Cycle	Removal Fuel Cycle	EFPY at Removal	Lead Factor	Fluence x 1E19 at Removal [n/cm ²]
Tested	S	0	1	1.26	3.44	0.293
	Y	0	5	5.86	3.43	1.07
	V	0	11	14.23	2.20	1.35
	B	5	25	34.87	2.65	3.62
Removed & Stored	T	0	5	5.86	3.43	1.07
	Z	0	5	5.86	3.43	1.07
	C	5	12	15.84	1.84	1.24
	D	5	12	15.84	1.84	1.24
Standby	A	5	in reactor	tbd	1.04	tbd
	U	0	in reactor	tbd	1.19	tbd
	X	0	in reactor	tbd	1.19	tbd
	W	0	in reactor	tbd	1.19	tbd

Notes: Red text indicates capsules containing samples of weld wire heat 27204.

Standby capsules remain in the reactor without a planned withdrawal date.

(Data from WCAP-13440)

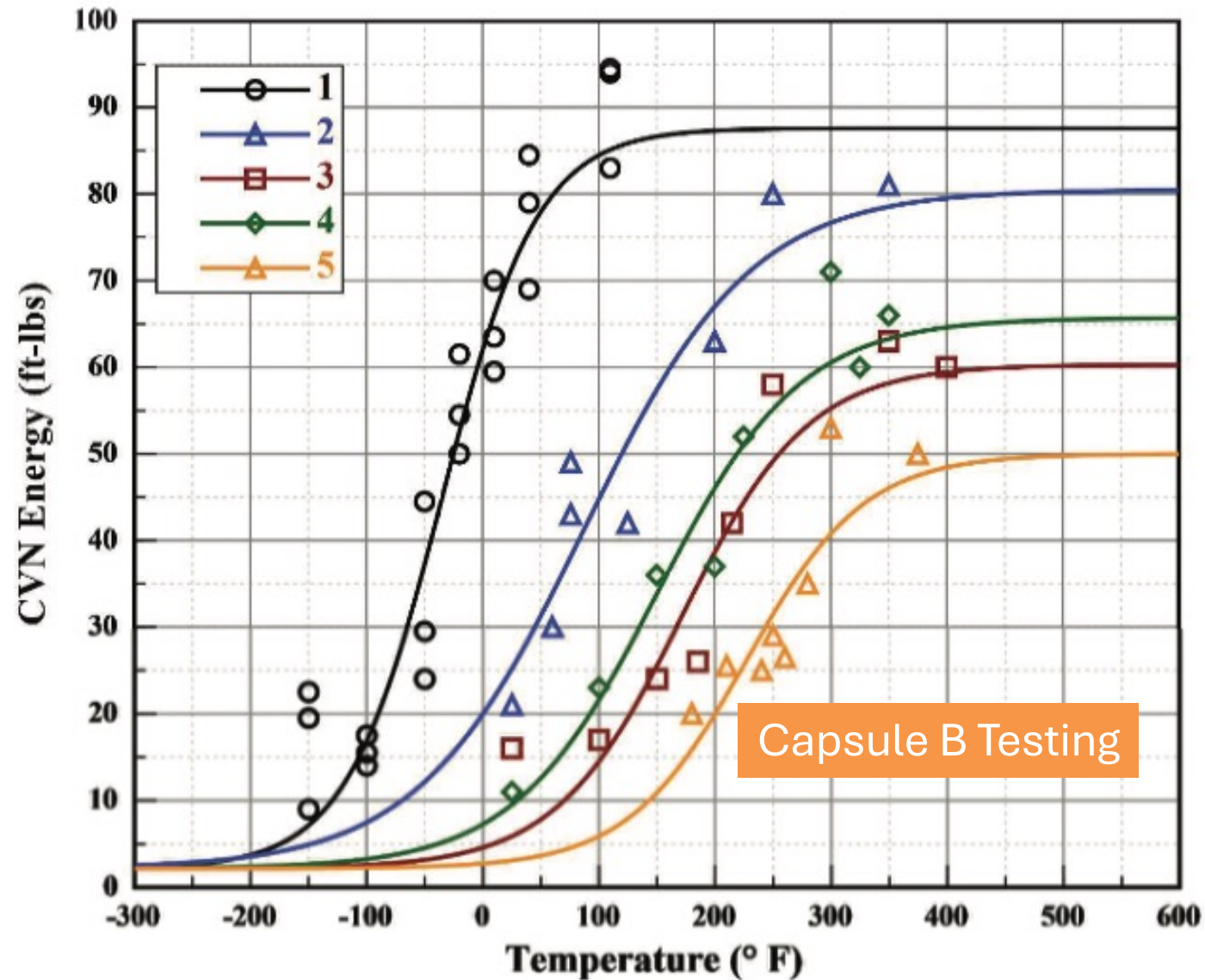
② Schematic of Fits to Charpy Data



② Data for Surveillance Weld

Curve ID

- 1: unirradiated
- 2: Capsule S
- 3: Capsule Y
- 4: Capsule V
- 5: Capsule B



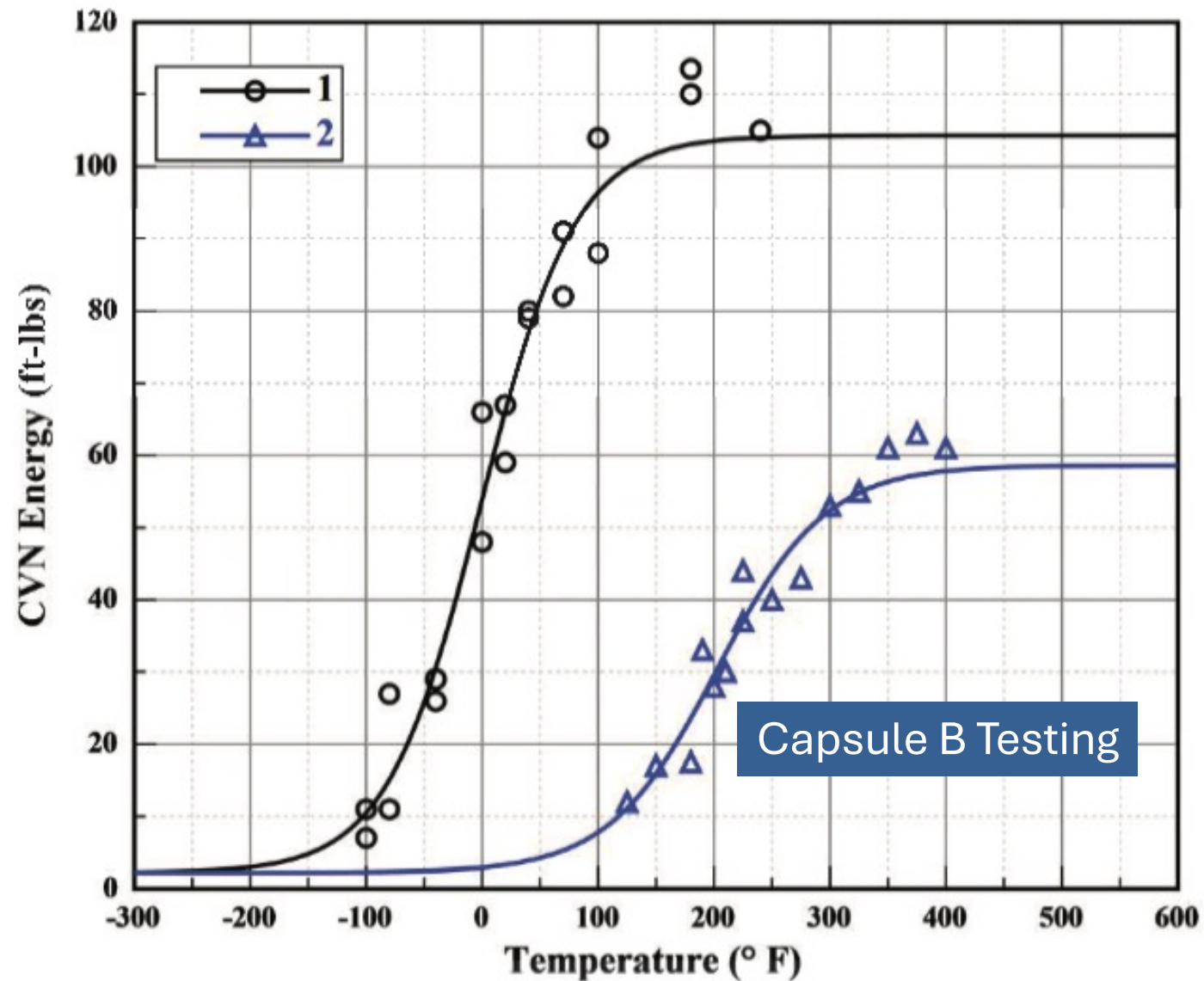
(Data from WCAP-19173)

② Data for Surrogate Weld

Curve ID

1: unirradiated

2: Capsule B



(Data from WCAP-19173)

② Surveillance Data for Heat 27204

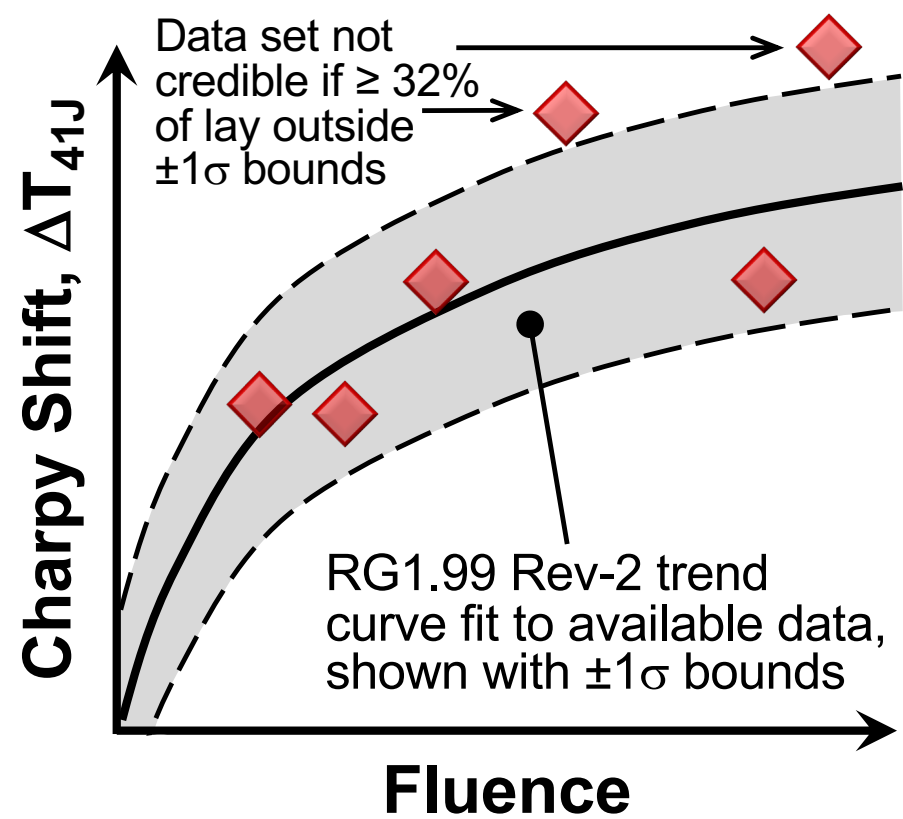
Plant	Capsule	Fluence ÷ 1×10^{19} n/cm ²	Time Averaged Cold Leg Temperature (°C)	T _{41J} (°C)	ΔT _{41J} (°C)	USE (J)	% USE drop	Cu (wt%)	Ni (wt%)
Diablo 1 (surveillance weld)	--	unirrad	--	-53.9	--	118.8	--	0.198	0.999
	S	0.293	284.4	+7.3	61.2	109.1	8.1		
	Y	1.07	283.1	+75.0	128.9	81.8	31.2		
	V	1.35	282.8	+57.5	111.2	89.5	24.6		
	B+S	3.91 ⁽¹⁾	282.0	+117.6	171.5	67.8 ⁽²⁾	42.9		
Diablo 1 (surrogate weld)	--	unirrad	--	-40.6	--	141.4	--	0.22	1.0
	B	3.62	282.0	+94.2	134.8	79.5	43.8		
Palisades (sister ⁷)	--	unirrad	--	-40.7	--	147.0	--	0.194	1.067
	SA-60	1.50	279.4	+99.9	140.6	71.9	51.1		
	SA-240	2.38	281.9	+108.1	148.8	72.9	50.4		

General Notes: Mean cold leg temperature for Diablo Canyon Unit 1 is 283.1 °C.
 Best estimate copper for weld 27204 is 0.2 weight percent
 Best estimate nickel for weld 27204 is 1.019 weight percent.
 Unirradiated RT_{NDT} used for Weld 27204 = -48.9 °C (generic value)

All data for weld wire heat 27204

3 Data Analysis: Credibility

What is credibility?	A credible data set follows RG predicted trends
	Not credible data deviates from these trends (see diagram)*



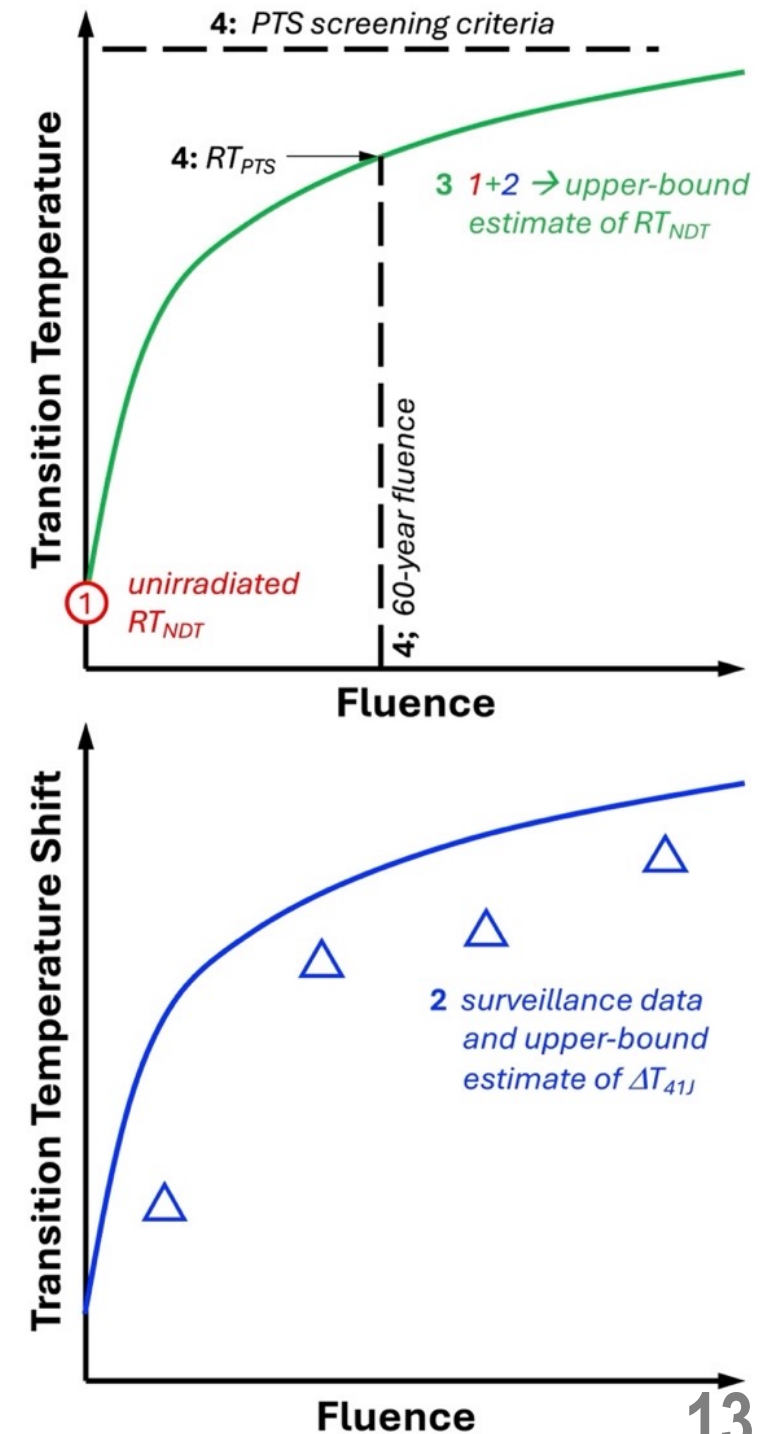
Total # of ΔT_{41J} Measurements	# of data allowed outside shaded bounds in a credible data set
2-3	0
4-6	1
7-9	2
10-12	3

* This is one of RG1.99's five credibility criteria. Other criteria, while checked, are typically met

RG1.99-R2 requires a not-credible data set to be modeled more conservatively, increasing the assessed embrittlement.

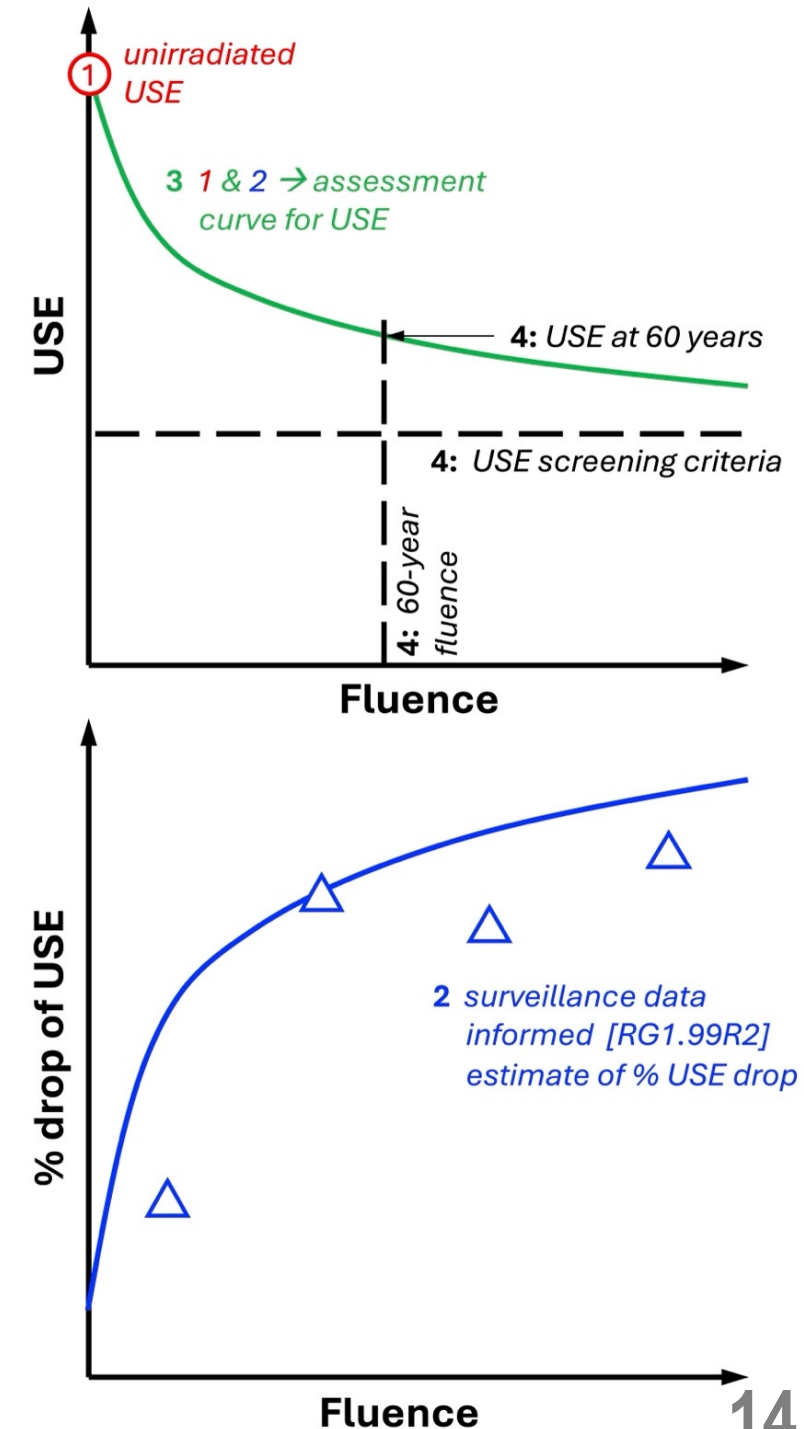
③ Data Analysis: PTS

1. Unirradiated value of RT_{NDT}
2. Assess surveillance data and credibility to establish the variation of an upper-bound estimate of ΔT_{41J} with fluence.
3. Add results from Step (1) & (2) to determine an upper-bound estimate of RT_{NDT} with fluence.
4. At the 60-year fluence, determine the value of the curve from Step (3), and compare to the NRC's PTS screening criteria



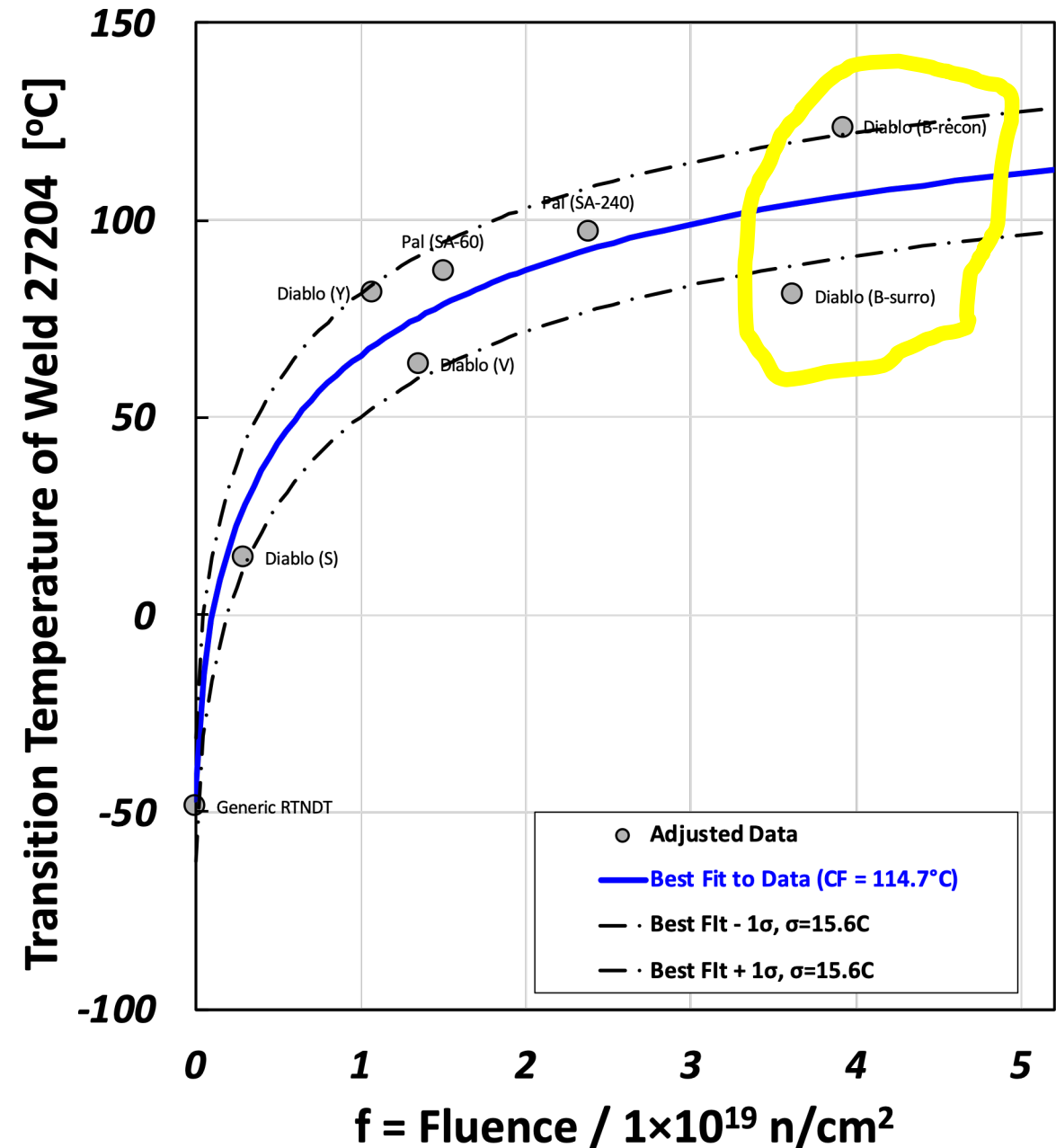
③ Data Analysis: USE

1. Obtain an unirradiated value of USE by Charpy testing
2. Assess surveillance data and credibility to establish the RG1.99-R2 informed estimate of %drop
3. Use the results from Step (1) and Step (2) to establish an assessment curve for USE.
4. At the 60-year fluence at the quarter-thickness ($\frac{1}{4}T$) location, determine the value of the curve from Step (3), and compare this value to the NRC's USE screening criteria



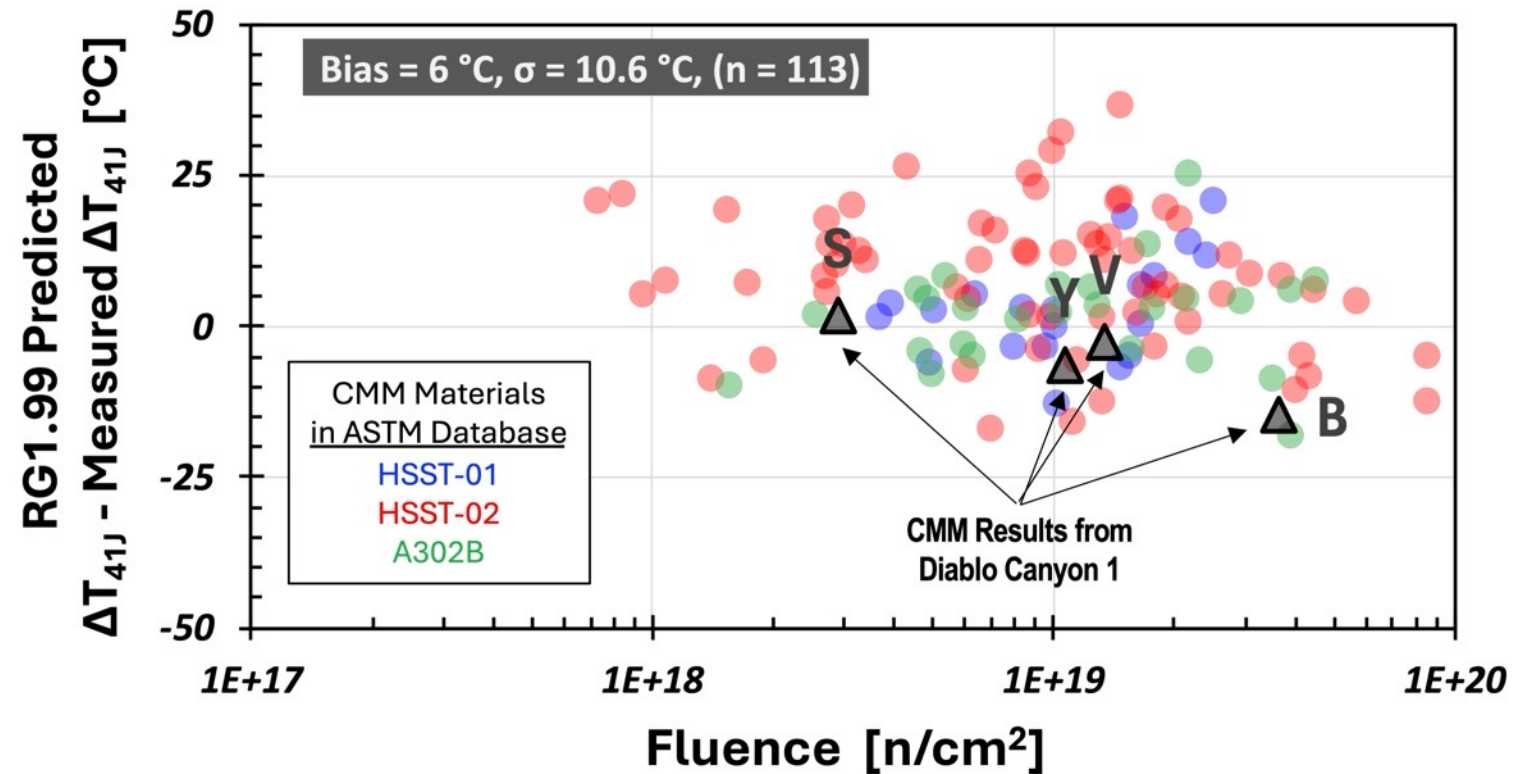
④ Assessment of Data Credibility

- Limiting material, weld wire heat 27204
 - 2 of 7 data lie further than $\pm 1\sigma$ from RG1.99 predicted mean curve
 - $2/7 = 28.6\%$
 - $28.6\% < 32\%$
 - Data set assesses as credible
- Comparison to Westinghouse result
 - Best fit CF matches to $< 1\%$ difference
 - Data set also assessed as credible



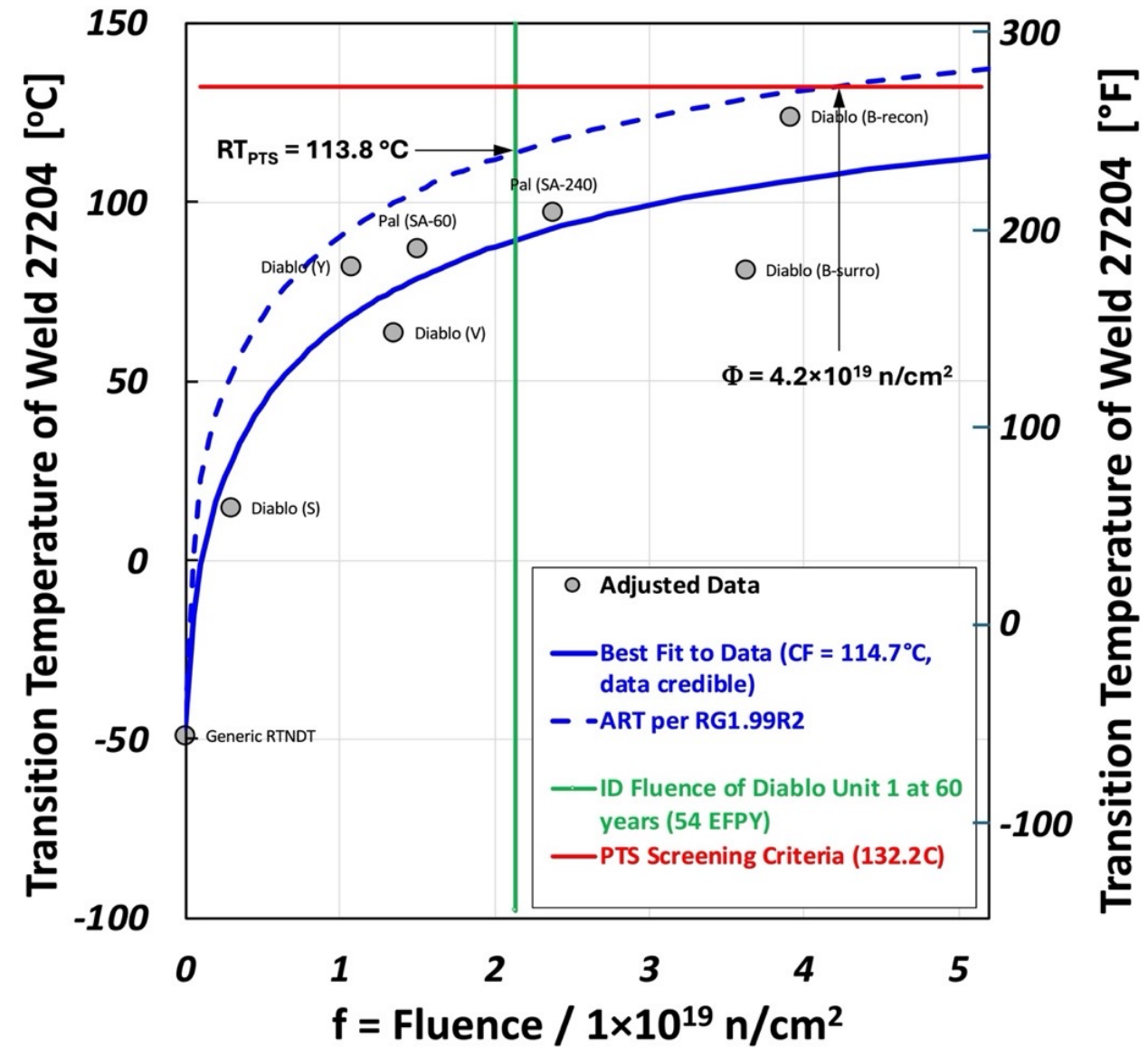
④ Assessment of Data Credibility

- Correlation monitor materials (CMM)
 - One material placed in many different reactors
 - Charpy data used to detect off-normal irradiation conditions
 - NRC requires CNN data from plant in question to “fall within the scatter band of the database” for the CMM material
 - The Unit 1 data (Δ) meet this criteria
 - Data set assessed as credible



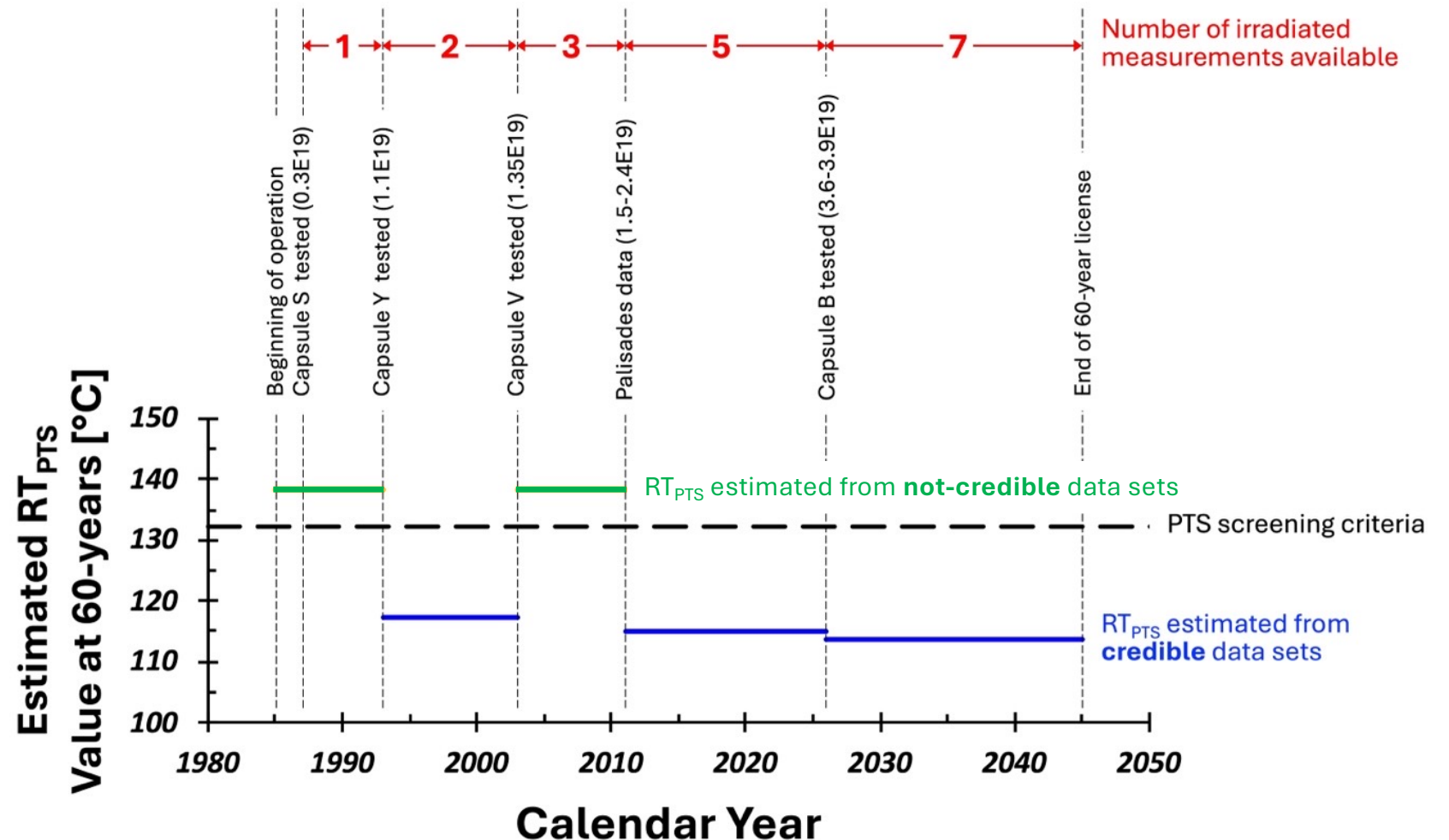
⑤ **PTS Screening Criteria: NRC Method**

- RT_{PTS} result
 - Value for Unit 1 is 113.8 °C at the 60-year fluence
 - This value is 18.4 °C below the NRC's screening limit
 - Result is acceptable
- Fluence at which Unit 1 will reach the screening criteria
 - Nearly twice that expected at end of licensed operation (60-years)



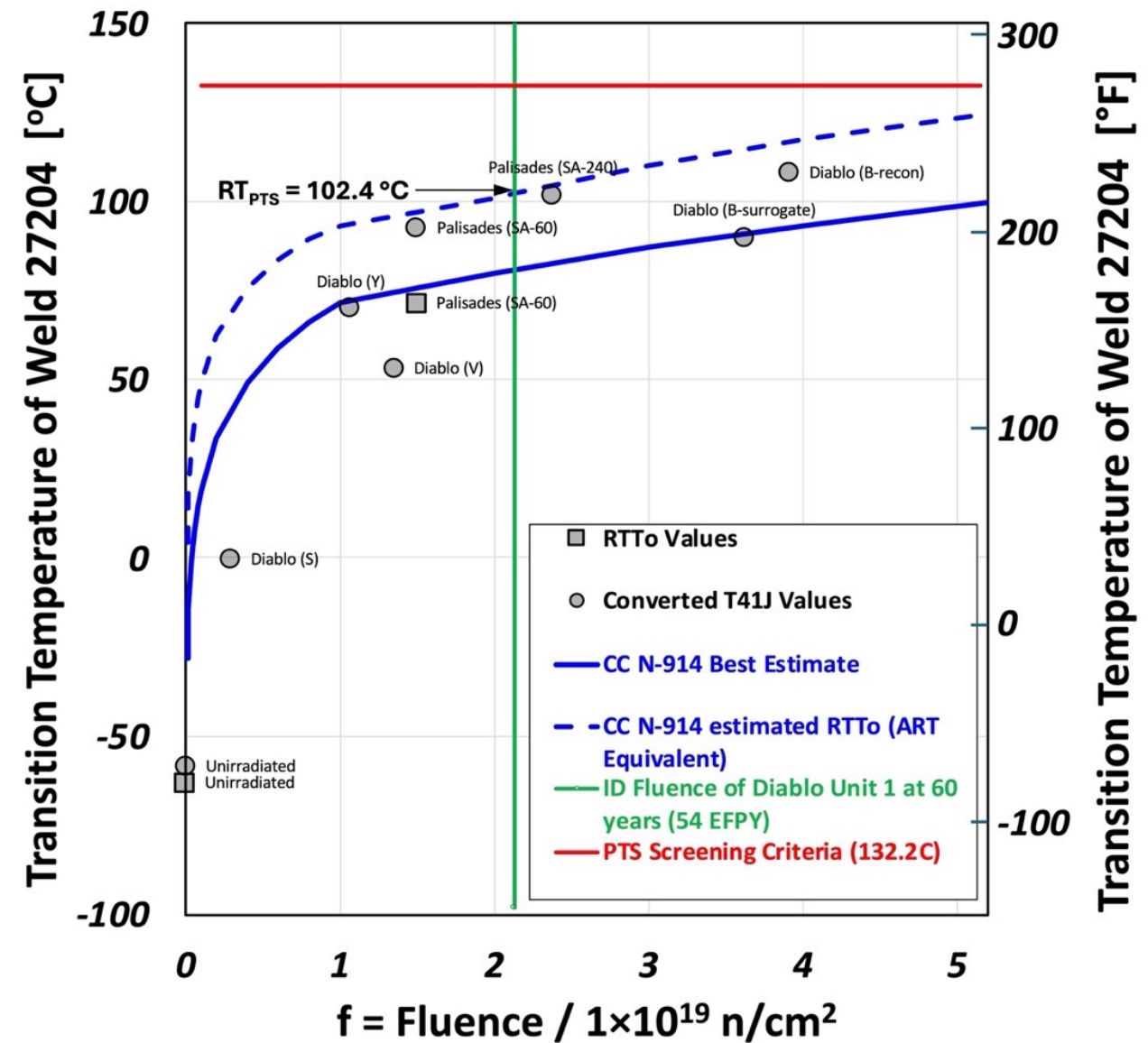
⑤ PTS Screening Criteria: NRC, Prior Results

- Consideration of Capsule B data reduces the prior RT_{PTS} estimate based on 5 data
- Estimated time to reach the PTS screening criteria
 - > 100 years from start of operation if data set is **credible** (current situation)
 - $\approx$ 50 years from start of operation if data set is **not-credible** (not the current situation)
- Increasing **data quantity** ($3 \rightarrow 5, 5 \rightarrow 7$) continues to confirm data credibility following NRC's criteria



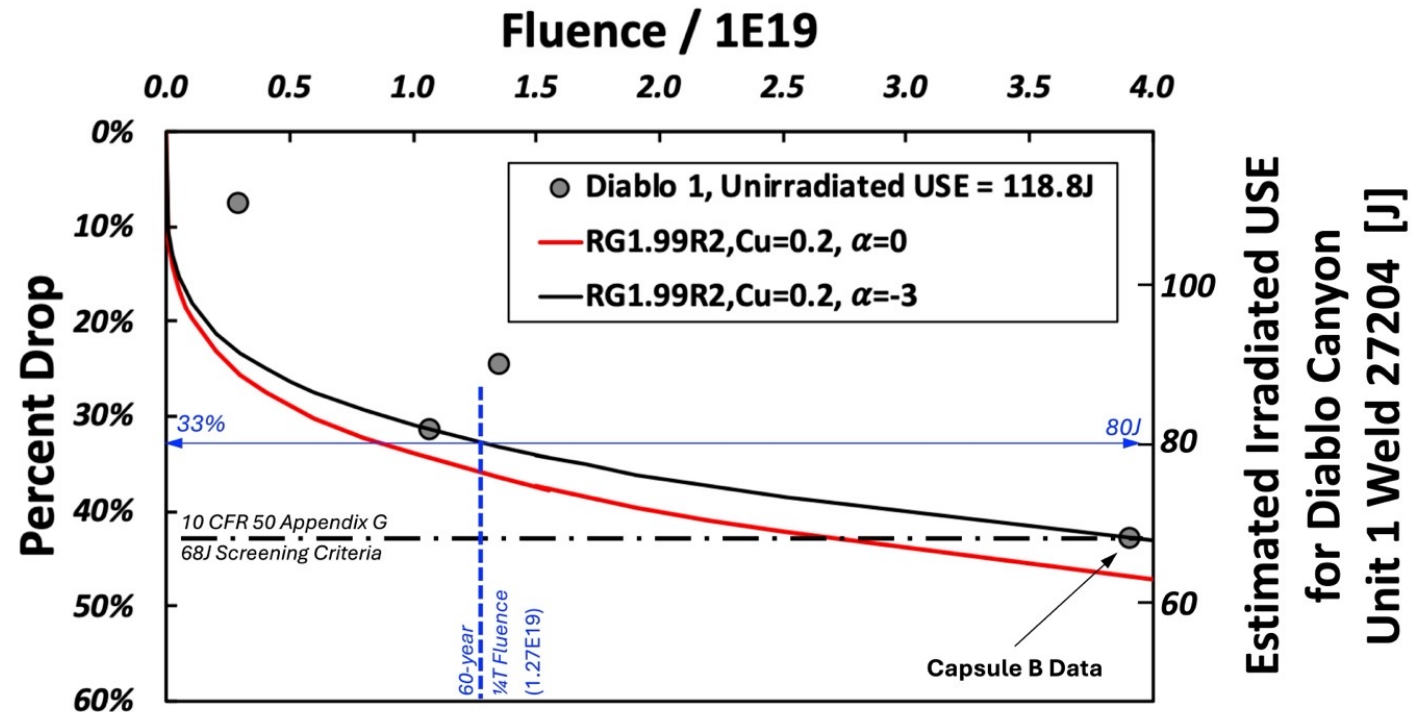
⑤ PTS Screening Criteria: Supplemental Method

- Supplemental method
 - Uses draft ASME approach (CC N-914)
 - Up-to-date method
 - Also uses fracture toughness data
 - Independent check on NRC approach
- RT_{PTS} result
 - Value for Unit 1 is 102.4 °C at the 60-year fluence
 - This value is 29.8 °C below the NRC's screening limit
 - Result is acceptable
- Fluence at which Unit 1 will reach the screening criteria
 - Nearly three times that expected at end of licensed operation (60-years)



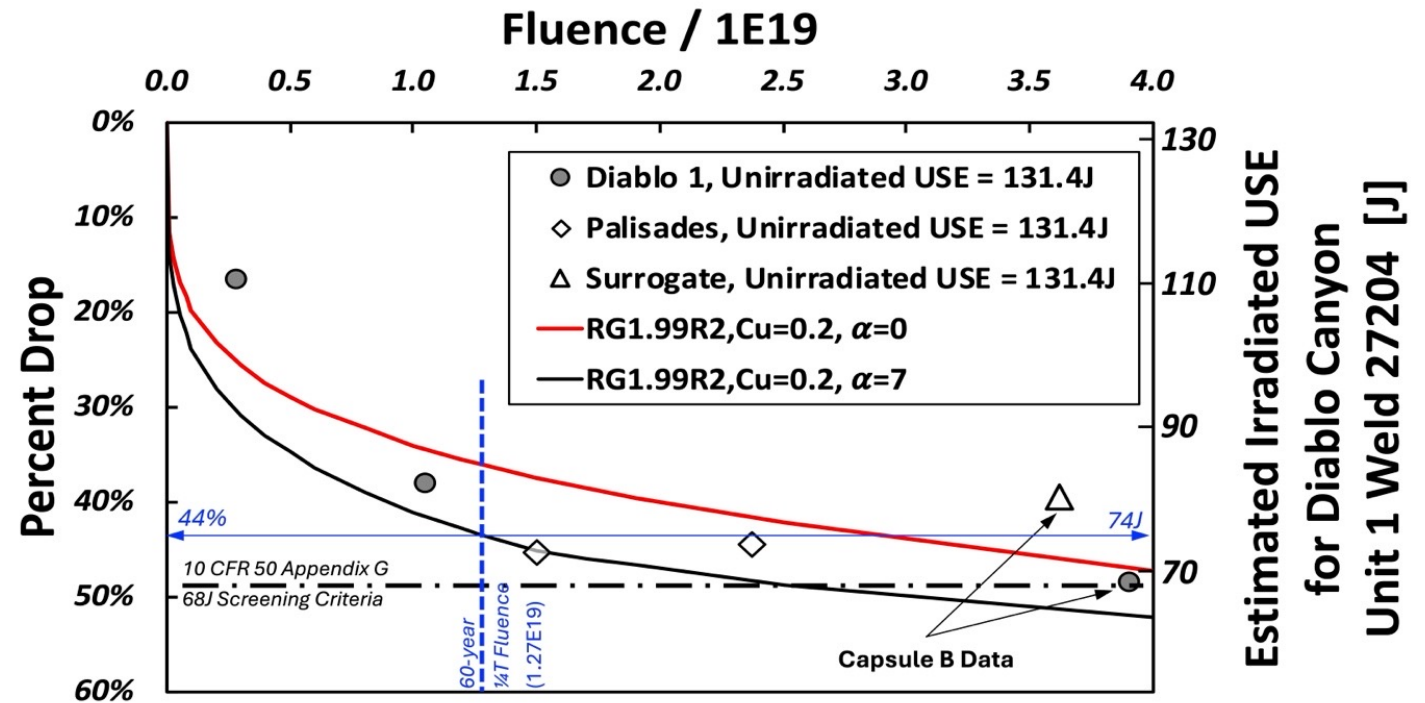
⑥ USE Screening Criteria: NRC Method

- USE result
 - Value for Unit 1 is 80J at the 60-year fluence
 - This value is 12J above the NRC's screening limit
 - Result is acceptable
- Fluence at which Unit 1 will reach the screening criteria
 - Over three-times that expected at end of licensed operation (60-years)



⑥ USE Screening Criteria: Supplemental Method

- Supplemental method uses all data for weld wire heat 27204, including data not required by the NRC
- USE result
 - Value for Unit 1 is 74J at the 60-year fluence
 - This value is 6J above the NRC's screening limit
 - Result is acceptable
- Fluence at which Unit 1 will reach the screening criteria
 - Approximately 1.8-times that expected at end of licensed operation (60-years)



④&⑤&⑥: NRC Review of Capsule B Report

- 24 September 2026
 - NRC issued staff assessment of the Capsule B report
 - ADAMS ML26264A267
- Outcome
 - **Credibility:** *“The staff noted that available surveillance data for the surveillance weld Heat #27204 is credible in accordance with Regulatory Guide 1.99 Revision 2”*
 - **PTS & P-T limits:** *“the staff noted that (1) the pressurized thermal shock reference temperature approved in the staff's evaluation (ML25153A508) of the DCP's license renewal application and (2) the adjusted reference temperature values used in the current DCP's Pressure and Temperature Limits Report (ML25210A382) continue to remain applicable when considering the surveillance data from Capsule B.”*
 - **USE:** *“the projected USE values approved in the staff's evaluation of the DCP's license renewal application continue to remain applicable when considering surveillance data from Capsule B”*

NRC's conclusions agree with those of Westinghouse/PG&E and with this consultant



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 2026

MEMORANDUM TO: Angela Buford, Branch Chief
Materials Branch 1
Division of Engineering
Office of Nuclear Reactor Regulation

FROM: On Yee, Senior Materials Engineer /RA/
Materials Branch 1
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: DIABLO CANYON POWER PLANT, UNIT 1 – STAFF ASSESSMENT
OF REACTOR VESSEL MATERIAL SURVEILLANCE CAPSULE
REPORT (CAPSULE B)

By letter dated June 29, 2026, (Agencywide Documents Access and Management System Package Accession No. ML26180A296), Pacific Gas and Electric Company (PG&E) (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a reactor vessel surveillance capsule report for Diablo Canyon Power Plant Unit 1 (DCPP, Unit 1) in accordance with the Title 10, Code of Federal Regulations (CFR), Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

The NRC staff reviewed the report for surveillance capsule B and finds that the report satisfies the requirements of 10 CFR Part 50, Appendix H. The NRC staff's assessment is enclosed.

Docket No. 50-275

CONTACT: On Yee, NRR/DE
(301) 415-1905

⑦ Possible Future Regulatory Changes & their Impact on Unit 1

- Rulemaking on embrittlement & surveillance
 - Requested by staff in 2022
 - Authorized in 2025, formal rulemaking not begun
 - Process takes 1-4 years to complete once begun
- Three topics
 - Embrittlement trend curve & assessment method
 - Surveillance monitoring requirements
 - Use of fracture toughness data

Rulemaking aims to update NRC guidance, last revised in the 1980s, making it more consistent with the current state-of-knowledge

The image shows a document from the U.S. Nuclear Regulatory Commission (NRC) titled "POLICY ISSUE (Notation Vote)". The document is dated March 8, 2022, and is identified by the number SECY-22-0019. It is marked as "OFFICIAL USE ONLY" and "SENSITIVE INTERNAL INFORMATION LIMITED TO THE NRC UNLESS THE COMMISSION DETERMINES OTHERWISE". The document is addressed to "The Commissioners" and is from "Daniel H. Dorman, Executive Director for Operations". The subject is "RULEMAKING PLAN FOR THE REVISION OF EMBRITTLEMENT AND SURVEILLANCE REQUIREMENTS FOR HIGH-FLUENCE NUCLEAR POWER PLANTS IN LONG-TERM OPERATION". The purpose is to request Commission approval to conduct rulemaking to amend the reactor pressure vessel (RPV) embrittlement and surveillance requirements in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," for high-fluence plants in long-term operation. The summary states that this paper provides the U.S. Nuclear Regulatory Commission (NRC) staff's request to initiate a rulemaking to revise the RPV embrittlement and surveillance requirements in 10 CFR Part 50. The rulemaking would revise Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 to include additional surveillance testing requirements for long-term operation and a revised fluence function fit (either a new embrittlement trend curve (ETC) or an update to existing trend curves) in the applicable regulations and implementing

CONTACTS: Stewart Schneider, NMSS/REFS
301-415-4123
David L. Rudland, NRR/DNRL
301-415-1896

Enclosure transmitted herewith contains Official Use Only - Sensitive Internal Information. When separated from Enclosure, this transmittal document is decontrolled.

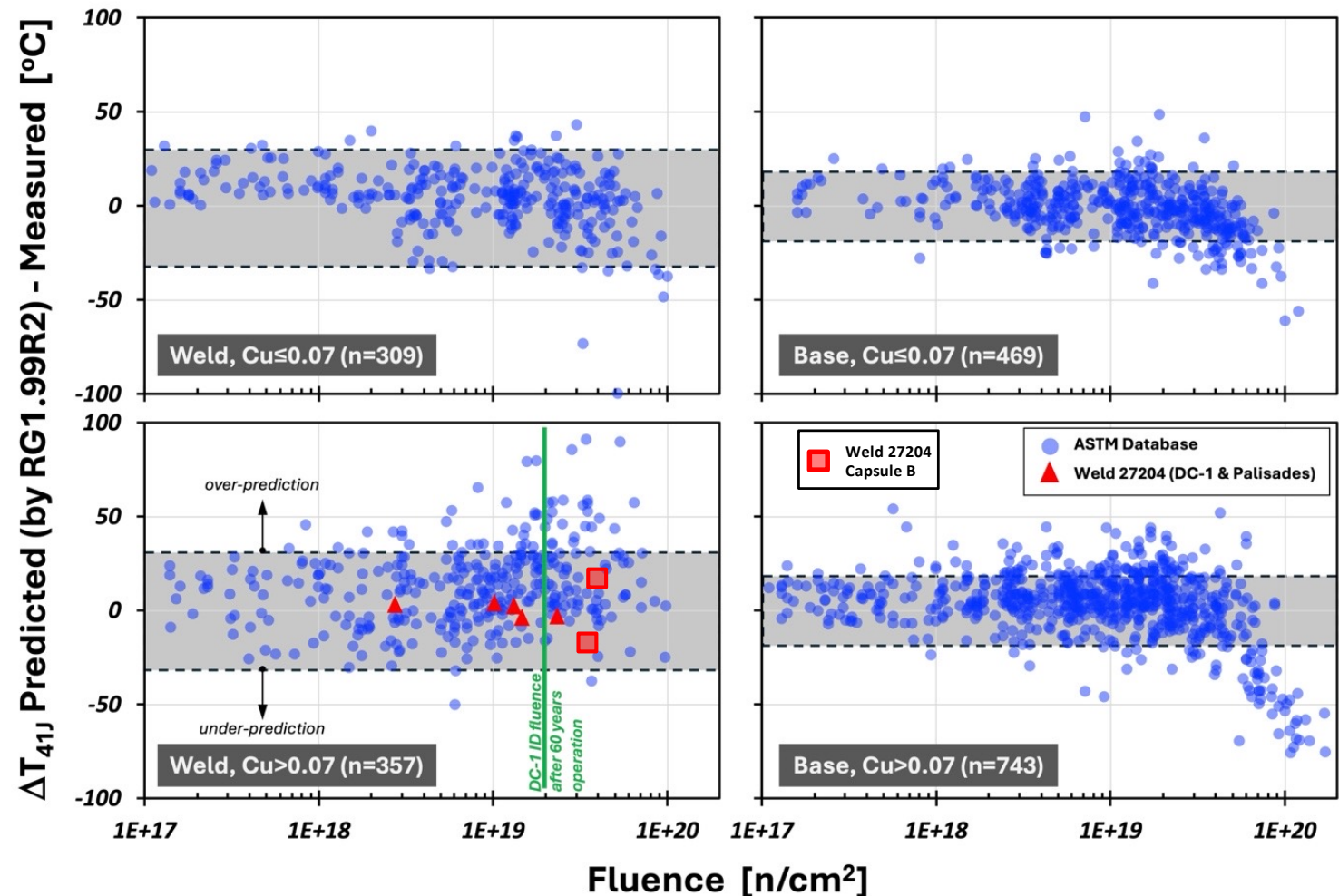
OFFICIAL USE ONLY
SENSITIVE INTERNAL INFORMATION
LIMITED TO THE NRC UNLESS THE
COMMISSION DETERMINES OTHERWISE

⑦ Embrittlement Trend Curve & Assessment Method

- Issue
 - NRC recognizes that their Reg Guide 1.99-Rev 2 trend curve does not model embrittlement trends well at high fluences ($>\approx 6 \times 10^{19}$ n/cm²)
 - Leads to under-prediction (non-conservative) of embrittlement at high fluence

Impact on Unit 1

- None
- Unit 1 60-year fluence is well below where NRC expresses concerns with RG1.99-R2



For graph, see Figure 3 from October 2024

<https://www.dciisc.org/download/library/topical-library/treatment-of-embrittlement-uncertainties-and-performance-monitoring-87.pdf>

⑦ Surveillance Monitoring Requirements

- Issue

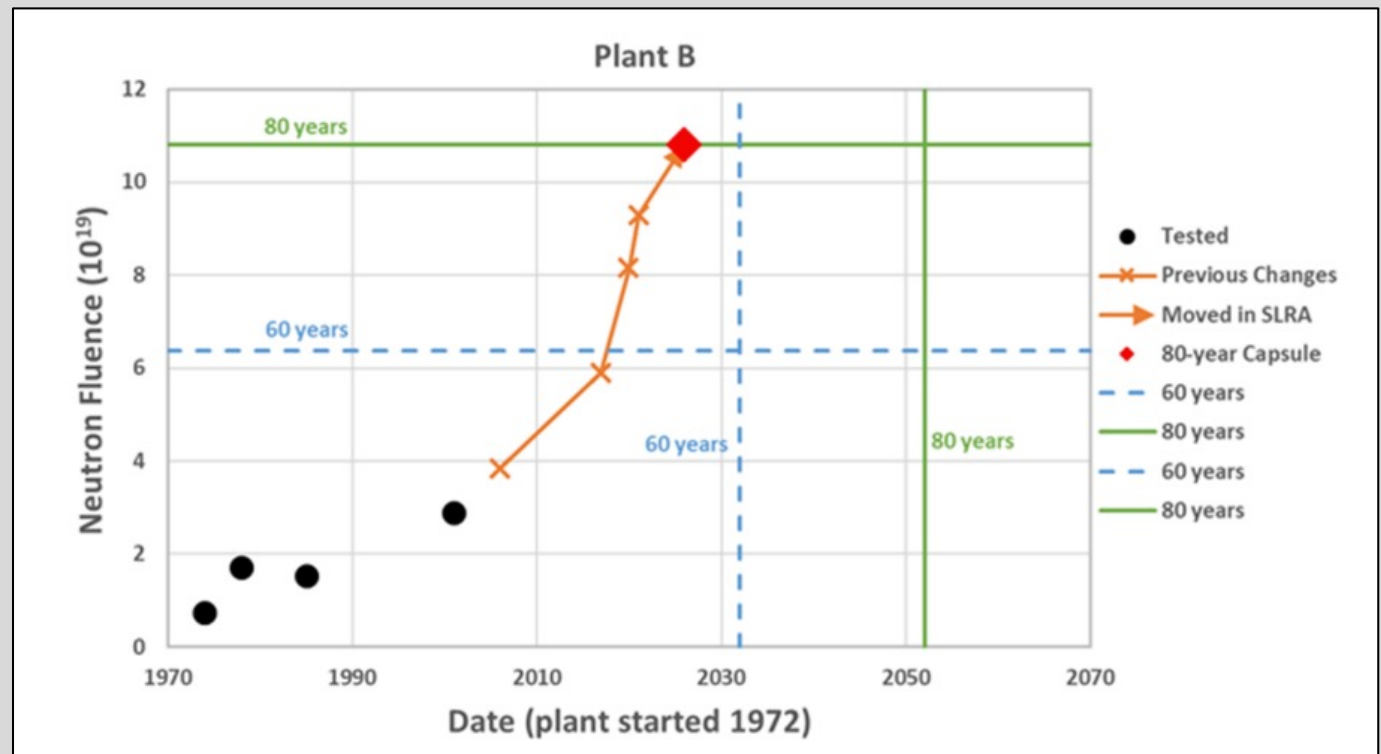
- NRC recognizes that their surveillance requirements in 10 CFR 50 Appendix H can lead to

- Long periods of no surveillance capsule testing
- Vessel currently having a fluence that exceeds that of the highest capsule

Impact on Unit 1

- Expected to be minimal
- Depends on how NRC re-writes their rules
- Unit 1 currently has data to nearly 2-times 60-year fluence

NRC Caption: Capsule withdrawal history for Plant B, where the withdraw of the capsule originally designed to apply to 40 years has been delayed multiple times, and is currently credited to address 80 years of plant operation, and the highest fluence data represent about 25 years of plant operation



For graph, see SECY-22-0019

<https://www.nrc.gov/docs/ML2131/ML21314A194.html>.

Note: The NRC's "Plant B" is not Diablo Canyon Unit 1

⑦ Allowing Use of Fracture Toughness Data

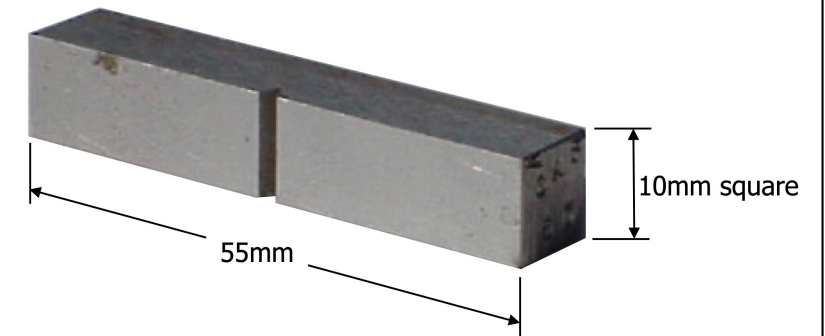
- Issue

- NRC requirements now depend entirely on Charpy and drop-weight specimens, which do not measure fracture toughness
- Fracture toughness measurement and assessment methodologies are well understood and standardized
- Ample case-specific experience exists
- NRC changes may make use of fracture toughness data by utilities more routine as an alternative method

Impact on Unit 1

- Positive impact
- The supplemental PTS analysis used fracture toughness data from heat 27204, demonstrating conservatism of current NRC approach applied to Unit 1

Charpy Specimen

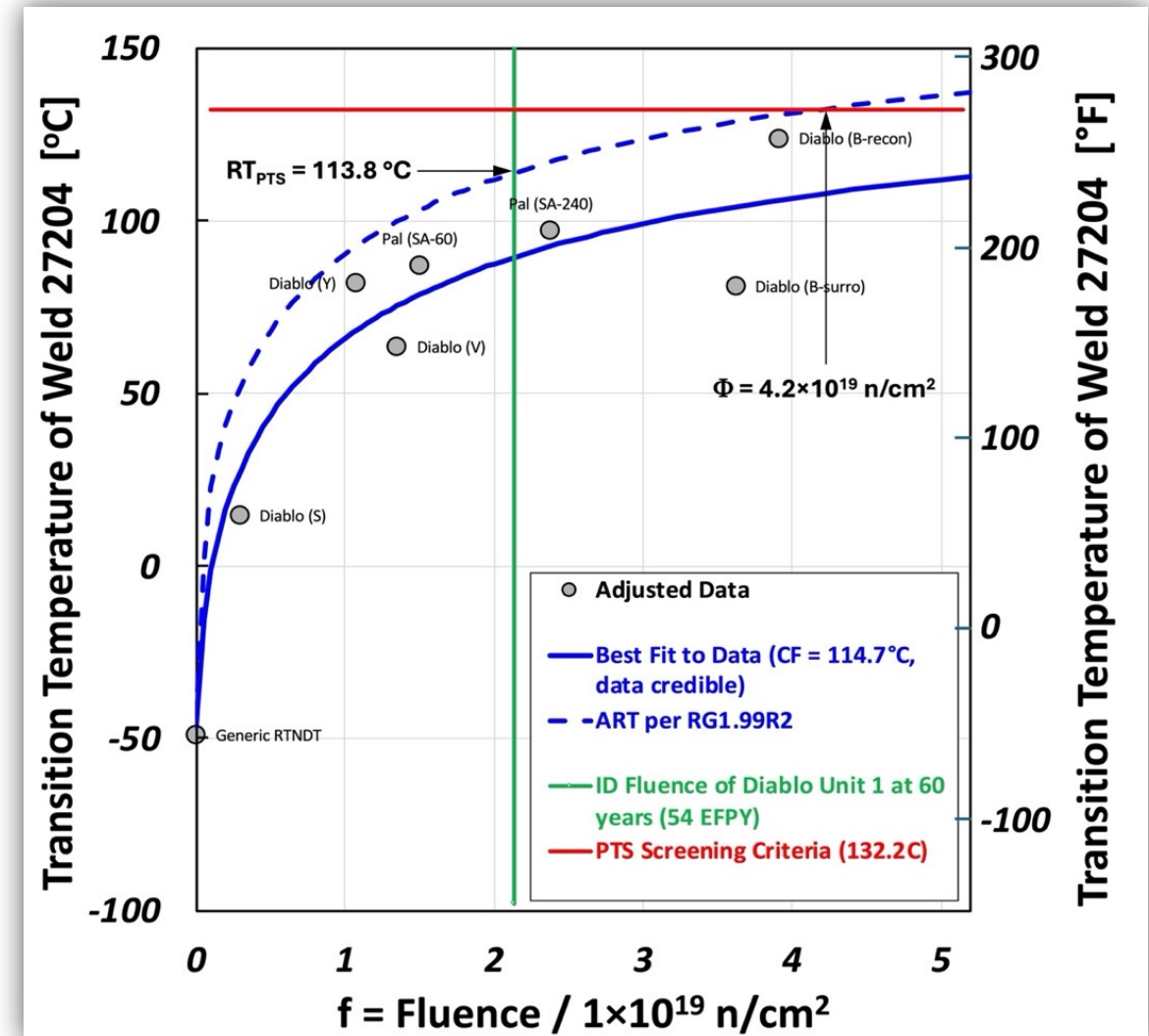


Mini-CT Fracture Toughness Specimen



⑧ Summary

- Impact of Capsule B results on Unit 1 embrittlement and safety assessments
 - Data set assessed as credible using NRC criteria
 - PTS assessment at 60 years
 - **Unit 1 complies with regulations**
 - 18 °C below screening criteria using NRC method
 - 30 °C below screening criteria using ASME CC N-914 method
 - USE assessment at 60 years
 - **Unit 1 complies with regulations**
 - 12J above screening criteria using NRC method
 - 6J above screening criteria considering supplemental data
 - Will not reach either screening criteria until decades after expiration of 60-year license
- NRC analysis agrees with these conclusions
- NRC rulemaking plans, once executed, are unlikely to negatively impact Unit 1



This assessment demonstrates that RPV embrittlement will not restrict the operation of Unit 1 through the end of its 60-year license.

Thank you for your time

Reports & Memos Written for DCISC

- Web Location
 - DCISC Topical Library:
<https://www.dcisc.org/resources/topical-library/>
 - Topic: Evaluation of Unit 1 Reactor Pressure Vessel Integrity
- 12 reports & memos are available on a range of topics
- Capsule B report appears on the public meeting page
 - https://www.dcisc.org/download/events/51_14-f-Diablo-Canyon-Independent-Safety-Committee-20261020-Capsule-B-Review-of-Data-Reported-in-WCAP-19173-Its-Implications.pdf

Evaluation of Unit 1 Reactor Pressure Vessel Integrity	
Letter Transmitting Two Reports by Dr. Mark Kirk to the DCISC. (PDF)	
Report Part 1: "Addressing Public Concerns." (PDF)	
Report Part 2: "Evaluation of Diablo Canyon Unit 1 Embrittlement." (PDF)	
Errata to Part 2 (PDF)	
Presentation Slides - Evaluation of Status of U-1 pressure vessel (PDF)	
Errata #2 to Part 2 (PDF)	
Dr. Kirk's Replies to Comments Made by Mr. Bruce Severance on April 15, 2024 (PDF)	
Replies to Dr Macdonald's Responses to Questions on his Evaluation of Dr Kirk's Reports (PDF)	
Replies to Mr Severance's Comments of July 18 2024 (PDF)	
Treatment of Embrittlement Uncertainties and Performance Monitoring (PDF)	
Assessment of Future Testing of Capsule B (PDF)	
Further Evaluation of Supplemental USE Analysis (PDF)	
Unit 1 Vessel Embrittlement Review of NRC 2025 SER (PDF)	
Memo: Reactor Vessel Embrittlement Regulations (PDF)	
Memo: 2022-2026 Comparison - Secy 22-0019 (PDF)	



Reactor Vessel Embrittlement Industry Comparison

- **Need Discussed During August 13 Briefing of Dr. Kirk with Dr. Meshkati**
- **Consultant McWhorter Agreed to Research Publicly Available Data**
- **All Plants Seeking Licensing Renewal Required to Submit Evaluations of Vessel Embrittlement as a Part of License Renewal Applications (LRAs)**
- **All NRC Safety Evaluation Reports (SERs) for LRAs Contain the NRC's Assessments of Each Plant's Vessel Embrittlement Evaluations Out to 60 Years of Operation (<https://www.nrc.gov/licensing-applications/status-initial-license-renewal-applications-and-industry-initiatives>)**
- **Standard LRA and SER Formats Place this Information into Section 4.2 of Each LRA and SER for a Majority of Plants**



Reactor Vessel Embrittlement Industry Comparison

- Information Contained in LRAs and SERs Typically Includes:
 - Surveillance Program Descriptions Including Vessel Embrittlement Limiting Plate or Weld Descriptions (including chemistry information) with Applicable RT_{PTS} Screening Criteria (270°F or 300°F)
 - Projected Effective Full Power Years (EFPY), Fluence, Upper Shelf Energy (USE), and RT_{PTS} at End of Renewal Period (60 Years)
 - Data Represents Licensing–Basis Projections for Initial License Renewals and Do Not Include Later Surveillance Data
- Sought to Gather Data from the Oldest Third of U.S. Pressurized Water Reactors (20 of 59) as Defined by Dates of License Expiration/Renewal
- Additional Sister Plants Easily Gathered (7) for a Total of 27 Reactors

Reactor Vessel Embrittlement
Industry Data - RTpts

Sorted by Degrees from Screening Criteria

Plant	60-Year End Date	60-Year EFPY	Fluence at 60-Year EFPY (x10 ¹⁹)	RTPTS at 60-Year EFPY	Screening Criteria	Degrees From Screening Criteria	Notes
Palisades (C)	2031	42	2.08	287	270	-17	1
Point Beach 2 (W)	2033	53	4.85	316	300	-16	2
Beaver Valley 1 (W)	2036	54	6.09	276	270	-6	2
Oconee 2 (B)	2033	48	1.20	300	300	-0.1	2, 4
Point Beach 1 (W)	2030	53	4.71	299	300	1	3
Surry 1 (W)	2032	48	5.40	269	270	1	
D. C. Cook 1 (W)	2034	48	2.83	283	300	17	
Arkansas Nuclear 1 (B)	2034	48	1.50	278	300	22	
Diablo Canyon 1 (W)	2044	54	2.13	246	270	24	
Robinson (W)	2030	50	2.45	275	300	25	4
St. Lucie 1 (C)	2036	54	3.08	241	270	29	
Ginna (W)	2029	54	5.01	271	300	29	
Three Mile Island 1 (B)	2034	52	1.97	264	300	36	
Oconee 1 (B)	2033	48	0.92	230	270	40	
North Anna 2 (W)	2040	52	5.91	228	270	42	4
D. C. Cook 2 (W)	2037	48	2.46	227	270	43	
Diablo Canyon 2 (W)	2045	54	2.45	225	270	45	
Oconee 3 (B)	2034	48	1.16	254	300	46	
Surry 2 (W)	2033	48	5.34	219	270	51	
North Anna 1 (W)	2038	50	5.90	191	270	79	4
Millstone 2 (C)	2035	54	4.05	191	270	79	
St. Lucie 2 (C)	2043	54	4.89	172	270	98	
Beaver Valley 2 (W)	2047	54	0.49	161	270	109	
Millstone 3 (W)	2045	54	3.31	135	270	135	
Prairie Island 1 (W)	2033	54	5.16	157	300	143	4
Arkansas Nuclear 2 (C)	2038	48	5.09	123	270	147	
Prairie Island 2 (W)	2034	54	5.19	136	300	164	

Notes

- Alternative Rule Used (10CFR50.61a)
Weld Heat W5214 Limiting (not 27204)
- Flux Reduction Program Implemented
- DCPP RT_{PTS} Data From 2025 NRC SER for LR
w/Cap. B, M. Kirk Calc New RT_{PTS} = 236.8 °F (113.8 °C)
= 33 °F from screening criteria
- Oconee fluences extrapolated from 54 EFPY data.

Reactor Vendor

- W Westinghouse
C Combustion Engineering
B Babcock & Wilcox

Plant	60-Year End Date	60-Year EFPY	Fluence at 60-Year EFPY (x10 ¹⁹)	RTPTS at 60-Year EFPY	Screening Criteria	Degrees From Screening Criteria	Note	Reactor Vessel Embrittlement Industry Data - RTpts
Beaver Valley 2 (W)	2047	54	0.49	161	270	109		Sorted by Fluence
Oconee 1 (B)	2033	48	0.92	230	270	40	1	
Oconee 3 (B)	2034	48	1.16	254	300	46	1	
Oconee 2 (B)	2033	48	1.20	300	300	-0.1	1	
Arkansas Nuclear 1 (B)	2034	48	1.50	278	300	22		
Three Mile Island 1 (B)	2034	52	1.97	264	300	36		
Palisades (C)	2031	42	2.08	287	270	-17		
Diablo Canyon 1 (W)	2044	54	2.13	246	270	24		
Diablo Canyon 2 (W)	2045	54	2.45	225	270	45		
Robinson (W)	2030	50	2.45	275	300	25		
D. C. Cook 2 (W)	2037	48	2.46	227	270	43		
D. C. Cook 1 (W)	2034	48	2.83	283	300	17		
St. Lucie 1 (C)	2036	54	3.08	241	270	29		
Millstone 3 (W)	2045	54	3.31	135	270	135		
Millstone 2 (C)	2035	54	4.05	191	270	79		
Point Beach 1 (W)	2030	53	4.71	299	300	1		
Point Beach 2 (W)	2033	53	4.85	316	300	-16		
St. Lucie 2 (C)	2043	54	4.89	172	270	98		
Ginna (W)	2029	54	5.01	271	300	29		
Arkansas Nuclear 2 (C)	2038	48	5.09	123	270	147		
Prairie Island 1 (W)	2033	54	5.16	157	300	143		Note
Prairie Island 2 (W)	2034	54	5.19	136	300	164		1. Oconee fluences extrapolated from 54 EFPY data.
Surry 2 (W)	2033	48	5.34	219	270	51		
Surry 1 (W)	2032	48	5.40	269	270	1		Reactor Vendor
North Anna 1 (W)	2038	50	5.90	191	270	79		W Westinghouse
North Anna 2 (W)	2040	52	5.91	228	270	42		C Combustion Engineering
Beaver Valley 1 (W)	2036	54	6.09	276	270	-6		B Babcock & Wilcox

								Reactor Vessel Embrittlement Industry Data - USE
Plant	60-Year End Date	60-Year EFPY	1/4t Fluence at 60-Year EFPY (x10 ¹⁹)	USE at 60-Year EFPY (ft-lbs)	Appendix G Value (ft-lbs)	Ft-lbs From Appendix G Value	Notes	
Oconee 2 (B)	2033	48	0.75	41	50	-9	1	Sorted by Ft-lbs From Appendix G Value
Arkansas Nuclear 1 (B)	2034	48	0.77	41	50	-9	1	
Surry 1 (W)	2032	48	2.87	42	50	-8	1	
Robinson (W)	2030	50	3.25	45	50	-5	1	
Oconee 1 (B)	2033	48	0.62	45	50	-5	1	
Oconee 3 (B)	2034	48	0.73	46	50	-4	1	
Palisades (C)	2031	42	N/A	49	50	-1	1, 3	
Surry 2 (W)	2033	48	0.66	49	50	-1	1	
Ginna (W)	2029	54	N/A	< 50	50	N/A	1, 2, 3	
Point Beach 1 (W)	2030	53	3.38	< 50	50	N/A	1, 2	
Point Beach 2 (W)	2033	53	3.46	< 50	50	N/A	1, 2	
Three Mile Island 1 (B)	2034	52	1.07	< 50	50	N/A	1, 2	
Beaver Valley 1 (W)	2036	54	3.80	52	50	2		
Diablo Canyon 2 (W)	2045	54	1.03	52	50	2		
Arkansas Nuclear 2 (C)	2038	48	3.02	54	50	4		<u>Notes</u> 1. Equivalent Margins Analysis (EMA) Performed and Found Acceptable 2. USE Data Not Provided in LRA Submitted as "Less than 50 ft-lbs" 3. 1/4t Fluence Data Not Provided in LRA or Subsequent Submittals
Millstone 2 (C)	2035	54	1.96	54	50	4		
North Anna 2 (W)	2040	52	3.68	56	50	6		
St. Lucie 1 (C)	2036	54	2.79	56	50	6		
North Anna 1 (W)	2038	50	3.68	57	50	7		
D. C. Cook 1 (W)	2034	48	1.70	57	50	7		
Diablo Canyon 1 (W)	2044	54	1.27	57	50	7		
Beaver Valley 2 (W)	2047	54	3.92	58	50	8		
Millstone 3 (W)	2045	54	1.97	59	50	9		
D. C. Cook 2 (W)	2037	48	1.48	66	50	16		
St. Lucie 2 (C)	2043	54	2.91	70	50	20		<u>Reactor Vendor</u> W Westinghouse C Combustion Engineering B Babcock & Wilcox
Prairie Island 1 (W)	2033	54	3.33	75	50	25		
Prairie Island 2 (W)	2034	54	1.67	84	50	34		



Reactor Vessel Embrittlement Industry Comparison

- **Conclusions:**
 - At the end of the renewal period (54 EFPY, 2044), Unit 1's projected RT_{PTS} of 246°F is 24°F below the 270°F screening criterion applicable to its limiting axial weld.
 - A preliminary recalculation (by Dr. Kirk) incorporating Capsule B data yields $RT_{PTS} = 236.8^\circ\text{F}$ (33.2°F below the screening criterion). This has not been submitted for NRC review. WCAP-18924-NP remains the analysis of record and the current licensing-basis margin remains 24°F.
 - At the end of the renewal period, Unit 1's projected USE is 57 ft-lbs, 7 ft-lbs above the 50 ft-lbs value required by 10 CFR 50 Appendix G (analysis of record). (Dr. Kirk calculation – 59 ft-lbs, 9 ft-lbs above.)



Reactor Vessel Embrittlement Industry Comparison

- **Conclusions (continued):**
 - **Unit 1's projected RT_{PTS} is mid-range among the oldest third of the U.S. PWR fleet, 9th of 27 by margin to criterion, 13th by RT_{PTS} . Eight surveyed units hold smaller margins, and four exceed their screening criteria and have taken additional NRC-reviewed actions (flux reduction at three and the alternate rule at Palisades).**
 - **Of the surveyed units, USE values for 12 of the 27 were projected below 50 ft-lb and have NRC-accepted Equivalent Margin Analyses. Unit 1's 7 ft-lb margin is mid-range among the 15 surveyed units above 50 ft-lb.**
 - **These comparisons address RT_{PTS} and USE against the 10 CFR 50.61 screening criteria and 10 CFR 50 Appendix G minimum value, respectively.**



Equivalent Margins Analyses

- 10 CFR 50 Appendix G requires that the predicted USE exceed 50 ft-lbs through end of license.
- If a plant has a USE predicted to fall below 50 ft-lbs it performs an “Equivalent Margins Analysis” (EMA) to assess its continued operating safety.
- EMAs are performed using elastic-plastic fracture mechanics using well-established procedures.
 - Regulatory Guide 1.161
 - Appendix K to Section XI of the ASME Code
- All plants that were predicted to fall below 50 ft-lbs have used EMAs to demonstrate the acceptability of their continued operating safety to the NRC.
- No EMAs were required for either DCCP unit.

Diablo Canyon

Independent Spent Fuel Storage Installation

Update on Multi-Purpose Canister Aging Management Inspections

Presented by:

Eric Brackeen, Manager, Dry Fuel Storage & Decommissioning Projects





Agenda

- Overview of Diablo Canyon (DC) Independent Spent Fuel Storage Installation (ISFSI)
- HI-STORM 100SA Design
- Multi-Purpose Canister (MPC) Aging Management Program (AMP)
- Aging Management Inspection Preliminary Results



Overview of DC ISFSI

- **Site-specific 10 CFR Part 72 License**

- Original 20-year license was issued by the Nuclear Regulatory Commission (NRC) on 3/22/2004.
- On 6/20/2025, the NRC issued the renewed license for the DC ISFSI to allow its operation through March 22, 2064.

- **Holtec HI-STORM 100SA system**

- Anchored cask design for seismic considerations.
- PG&E has completed 8 loading campaigns and has 70 storage casks in service.



HI-STORM 100SA Design

Scale Model

- **MPC – houses the fuel (up to 32 assemblies)**
 - Integrally-welded pressure vessel
 - Includes an outer shell (0.5” thick), a lid (9.5” thick), and a base plate (2.5” thick)
 - Sits on pedestal within the overpack (fabricated as a metal-encased concrete disc, up to 16” thick)
- **Overpack – houses the MPC**
 - Made of 1” thick inner and outer concentric metal shells
 - provides approximately 30” of annular space that is filled with high density concrete
 - Includes a baseplate (2” thick), and a bolted lid (fabricated as a metal-encased concrete disc, up to 19” thick)

Multi-Purpose Canister (MPC)



MPC sits on pedestal

Overpack



MPC Aging Management Program

- Requires visual inspection of the accessible external surface areas of the MPC within the overpack.
- A condition monitoring program, implemented through inspection and trending, that ensures safety functions of the MPC are maintained.
 - Confinement
 - Structural Integrity
 - Radiation Shielding
 - Heat Transfer Functions
- Consistent with American Society of Mechanical Engineers (ASME*) Code Case N-860 and has been approved by the NRC.
 - Determines the site and MPC susceptibility to stress corrosion cracking
 - Performs screening exam, assessment exam (if required)
 - Sets scope and inspection frequency based on results and susceptibility
- A minimum of two (2) MPCs must be inspected at the DC ISFSI based on EPRI ISFSI Susceptibility Assessment Criteria (Technical Report 3002005371, Chapter 3).

*ASME is a not-for-profit membership organization that is known for setting codes and standards for mechanical applications. ASME conducts one of the world's largest technical publishing operations and holds numerous technical conferences and hundreds of professional development courses each year.

MPC Inspection Process

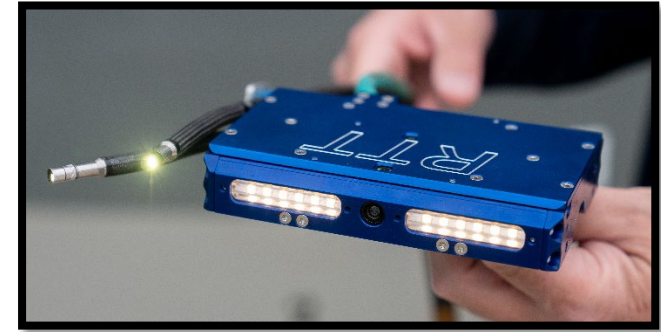
- **Accessibility**

- MPCs are accessible for inspection using remote robotic systems only.
- Robotic system is deployed into the HI-STORM outlet vents and secures to the inner wall of the overpack.

- **Qualification**

- Remote inspections of the MPCs are performed per ASME Code Section XI, for VT-3 and VT-1 visual examinations.
 - Qualified and Calibrated Equipment
 - Certified Examiners

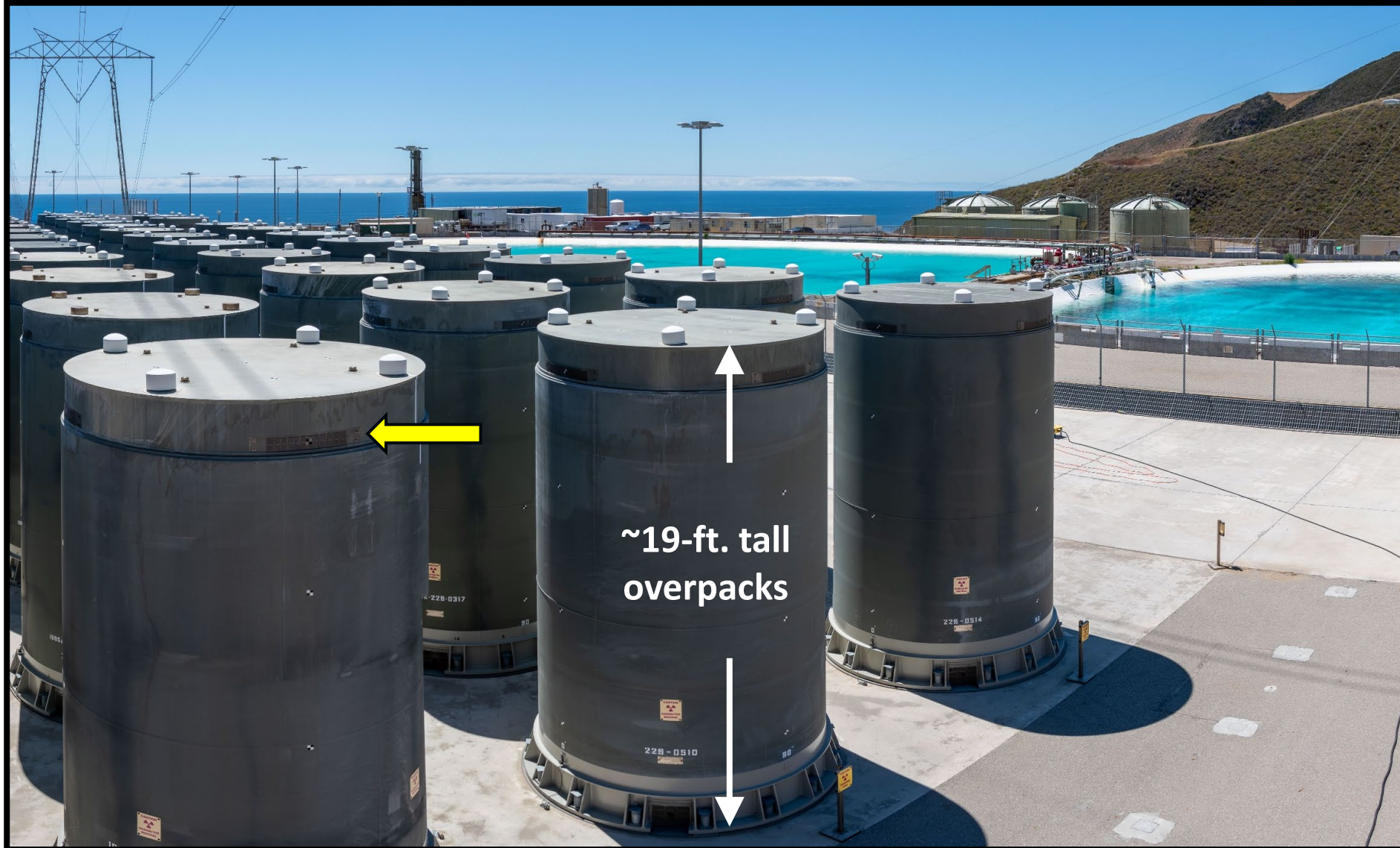
RTT Magnetic Crawler System



Inspection Set-up for Crawler System



MPC Inspection Process



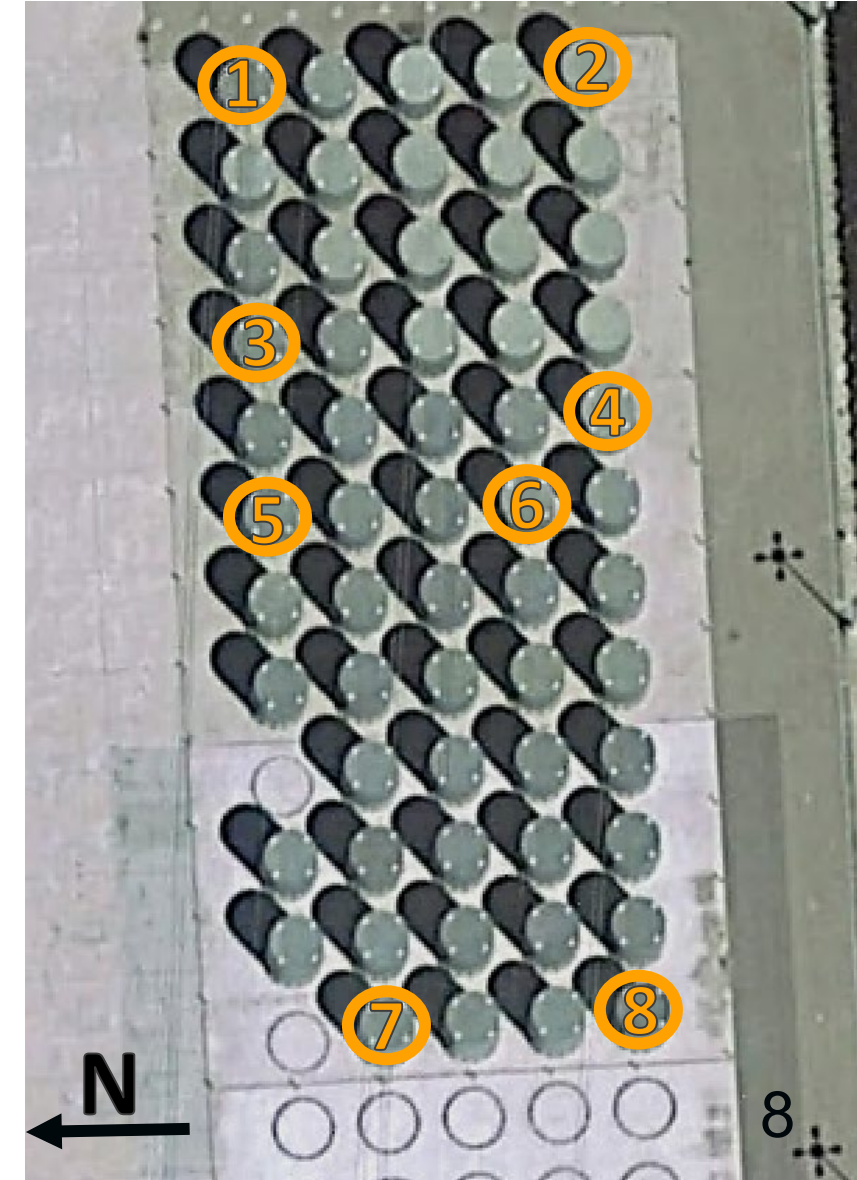
MPC Aging Management Inspections

• Scope

- Eight (8) MPCs and overpack interiors: same number and locations from the 2021 pre-application, baseline inspections
- Temperature Measurements
- Surface Cleaning Demonstration
- Exceeded regulatory requirements

• Inspection Process

- Conduct VT-3 exam for all accessible MPC and overpack interior surfaces
- Progress to a more detailed exam (VT-1) for new or changed service-induced anomalies (i.e., characterize)
- Clean surface if exams cannot determine metal condition under the deposits; re-examine cleaned area





MPC Aging Management Inspections

- **Preliminary Results**

- Inspections conducted by PG&E, Holtec, and RTT (robotic crawler vendor) the weeks of August 17th and 24th
- MPCs and overpacks continue to safely perform their intended functions
- No changes to inspected surface condition since 2021; no SCC indications present
 - Further validated with VT-1 exams in 2026 – provided greater detail of anomalies
 - Surface cleaning completed on representative area – confirmed rust-colored indications were superficial (i.e., do not penetrate the base metal)
- Based on initial results, no immediate or interim actions required

- **Next Steps**

- Accept the vendor inspection report (ECD Q4 2026)
- Use the Corrective Action Program to document an engineering evaluation of the inspection results (ECD Q1 2027), including:
 - Determination if any adjustments are need to the inspection frequency, methods, or scope
- Continue conducting periodic inspections: next inspection currently set for 2031 (every five years: licensing requirement)

Thank You

Eric Brackeen
Diablo Canyon Power Plant
Manager, Dry Fuel Storage & Decommissioning Projects





Dr. Raluca Scarlat, Member, with Richard St. Onge, and Consultants

1. Reactor Vessel Embrittlement Sample Capsule B Report
2. Independent Spent Fuel Storage Installation (ISFSI) Spent Fuel Canister Inspections
3. Westinghouse Issuance of 10 CFR Part 21 Reporting of Defects and Noncompliance
4. Spent Fuel Pool Arrangement and Expedited Transfer of Spent Fuel to the ISFSI
5. Meet with Nuclear Regulatory Commission (NRC) Resident Inspector
6. Meet with DCCP Officer
7. Flow Accelerated Corrosion
8. Station Update
9. 2026 Evaluated Emergency Preparedness Drill
10. Emergency Preparedness Meetings
11. Offsite Emergency Protective Action Measures
12. Public Outreach-Interview with Local Radio Station



Reactor Vessel Embrittlement Sample Capsule B Report

- On June 29 and July 1, 2026, Dr. Scarlat and Consultants met to receive a brief from PG&E on the Unit 1 Reactor Vessel Embrittlement Sample Capsule B. Others were in the briefing.
- Discussed subjects included:
 - Numbers and composition of material samples tested and the test results
 - Dosimetry (fluence) and thermal data for the sample capsule
 - Data creditability per Regulatory Guide 1.99
 - Quality assurance/quality control (QA/QC) and metrology for the temperature, dose, and mechanical property testing
 - Independent review of data post processing and interpretation
 - Quality assurance/quality control (QA/QC) and metrology for the temperature, dose, and mechanical property testing



Reactor Vessel Embrittlement Sample Capsule B Report (continued)

Conclusion: DCPD presented the results of testing and evaluation of material specimens contained in Reactor Vessel Embrittlement Sample Capsule B, which was removed from the vessel on April 21, 2025. PG&E concluded that the results confirmed that Unit 1 continues to meet NRC requirements for Reactor Vessel integrity and remains safe for at least 60 years of operation, and made its submission to the NRC on June 29, 2026, 4 months ahead of the 18-month regulatory schedule. The NRC staff concluded its analysis of the Capsule B report and concurred with its conclusions, on Sept. 24th, 2026. Capsule B results and analysis and the NRC staff analysis are publicly available.



Reactor Vessel Embrittlement Sample Capsule B Report (continued)

Recommendations to the DCISC: The DCISC should initiate and issue an explanation for an audience of general public and public officials, written by a panel of three to five experts providing multiple perspectives and spanning the disciplines of embrittlement, post-irradiation examination, non-destructive evaluation, neutron dosimetry, reactor safety, quality assurance, and risk communication, evaluating and describing for transparency to the public the following:

1. The pedigree of the data and of the analysis provided by Westinghouse, which was the basis for PG&E conclusions with regards to U1 embrittlement, including verification and certification processes, quality assurance and quality control; and
2. The type of decisions that are informed by capsule embrittlement data to improve confidence in plant safety, including but not limited to: the role of standby capsules, coupon reconstitution for fracture toughness measurements, inspections and inspection intervals, RPT limits, annealing, and probabilistic fracture toughness analysis. (D9/26FF-3.1A)

(continued)



Reactor Vessel Embrittlement Sample Capsule B Report (continued)

Recommendations to the DCISC: The DCISC should follow-up on quality assurance oversight of vendor-supplied safety-related data and analysis, including but not limited to the results of the ongoing PG&E review of Westinghouse's Quality Assurance Program and of the upcoming Nuclear Procurement Issues Committee (NUPIC) audit. (D9/26FF-3.1B)

Recommendation to PG&E: The following previous recommendation to PG&E was reviewed in this Fact-Finding Meeting and should be closed: *Recognizing that Capsule B test results are not necessary to assure Unit 1 Reactor Pressure Vessel embrittlement acceptability, the DCISC recommends that the Capsule B be removed at the next outage in 2025 and that a re-evaluation of Reactor Vessel integrity, including both PTS and USE, be performed once the new data from Capsule B become available.* (AR23/24AR23-2)



Independent Spent Fuel Storage Installation (ISFSI) Spent Fuel Canister Inspections

- The Fact-Finding Meeting (FFM) was held on August 18 and 19, 2026 at the DCPD station where the ISFSI Canister inspections were taking place.
- Inspections of a sample of the spent fuel canisters is an NRC requirement of the Aging Management Program (AMP) for extended operations under the Diablo Canyon renewed ISFSI License SNM-2511.
- Eight Multi-Purpose Canisters (MPC) were inspected by Non-Destructive Examination (NDE) techniques, conducted by Technicians Certified in VT 1, and VT 3 visual Inspections.
- Each MPC is prepared for inspection by removing the cooling air outlet screen and accessing the gap between the canister and the overpack which shields the spent fuel to minimize radiation.
- The two certified visual inspectors were both viewing the camera feed from the gap between the inner overpack and the outer canister surface. One inspector was stationed at the air vent with a smaller camera screen; the other inspector was viewing screens inside the viewing tent.



ISFSI Spent Fuel Canister Inspections (continued)

- Our allowable occupational dose for the planned observation activities was 10 mrem cumulative dose. Our exposure for our two day stay with the ISFSI inspection team was under 1 millirem.
- The inspections were slow and methodical, as expected there were a few signs of machining marks on the stainless-steel outer canister walls.
- The FFM Team returned to the DCPD location on Sept 1, 2026 to get an update on the remaining canisters to be inspected.
- One canister, #170, had pseudo rust spots along the axial and circumferential welds. These spots were brushed away with a small emery pad and quickly dissipated to reveal clean stainless steel.
- It appears that during manufacturing the canister, final structural welds were brushed with a carbon steel brush, not a stainless-steel brush to remove welding slag & debris.



ISFSI Spent Fuel Canister Inspections (continued)

Conclusion: Preparations for the Independent Spent Fuel Storage Installation (ISFSI) Canister Inspections were well planned. The execution phase of inspecting the canisters and overpack was performed in an excellent and professional manner. The successful removal of the corrosion-like spots from the surface of the canister welds revealed no degradation beneath the removed rust spots. It is a notable finding that the corrosion deposits observed can be cleaned off using the robot with attached emery cloth. The final ISFSI inspection report is being prepared by PG&E.

Recommendation to the DCISC: When the ISFSI final inspection report is available, DCISC should review the observations and conclusions recorded in the report, including comparisons with the prior inspection report, and the recommendations that will be made by the report with regards to whether changes are needed to future inspection plans, and including conclusions about the use of the emery cloth robotic arm. (D9/26FF3.2)



ISFSI Spent Fuel Canister Inspections (continued)



Dr. Scarlat



Westinghouse Issuance of 10 CFR Part 21 Reporting of Defects and Noncompliance

- Westinghouse issued a Part 21 regarding a “discovery of a deviation in its fuel data management system that has potential to affect the criticality calculations for fresh fuel with Integral Fuel Burnable Absorbers (IFBAs) in the spent fuel pool”.
- The Fact-Finding Team met with a DCPD Senior Reactor Engineering, Subject Matter Expert (SME).
- The SME verified the new fuel in the spent fuel pool does contain integral fuel burnable absorbers (IFBAs) as part of DCPD's fuel management process. New fuel rods are sprayed with IFBA containing chemicals that reduce the chances of spent fuel pool criticality.
- The SME explained that the amount of IFBAs associated with the new fuel bundles at DCPD contained twice (or more) times the standard amount of IFBAs for a typical Westinghouse reactor.



Westinghouse Issuance of 10 CFR Part 21 Reporting of Defects and Noncompliance

(continued)

- DCPD has new fuel bundles that have significant IFBA on the new rods that the reactor engineers use to plan storage of the new bundles in the spent fuel pool, prior to refueling.
- The Part 21 potentially affected 11 PWR sites, however some, but not all of the sites may be subject to the Part 21 issue as they may have new fuel rods with minimal IFBA applied to these fuel rods.
- The SME was well prepared for his meeting with the DCISC FFT. His handouts, such as the applicable DCPD Technical Specification for storing new fuel in the vicinity of spent fuel, helped the DCISC Team to better understand the level of analysis that is performed in fuel management preparation that support a refueling outage.

(continued) 11



Westinghouse Issuance of 10 CFR Part 21 Reporting of Defects and Noncompliance

(continued)

Conclusions: PG&E provided a satisfactory update regarding their response to the issuance of a 10 CFR Part 21 Notification by Westinghouse. The Reactor Engineering staff at Diablo Canyon maintained an additional amount of neutron absorbers, well beyond the minimum amount necessary to store new fuel in the Spent Fuel Pool prior to the next Unit 1 outage. The Part 21 issue continued to be pursued by the Westinghouse fleet, and the subject deviation in its fuel data management system was being examined. Effects are correctly managed by PG&E following the Technical Specifications and racking strategy, independently of the TrackWorks system.



Spent Fuel Pool Arrangement and Expedited Transfer of Spent Fuel to the ISFSI

- An email communication was sent to the Diablo Canyon Independent Safety Commission (DCISC) on Friday, June 5, 2026 by Dr. Frank N. von Hippel, from Princeton University.
- Dr. von Hippel recommended that Diablo Canyon should “re-rack the spent fuel pools to their original “open-rack” designs in which the spent fuel assemblies were separated such that no criticality will occur.
- Additionally, Dr. von Hippel recommended expedited transfer of the spent fuel to the ISFSI as quickly as possible, generally after 5 years of cooling time, to reduce the risk of spent fuel fire.
- DCISC requested a position paper from DCCP on Dr. von Hippel’s recommendations.
- DCCP provided a detailed response to Dr. von Hippel’s recommendations. The detailed Request and the Response are contained in the Sept FFM, Enclosure 2 and 3.



Spent Fuel Pool Arrangement and Expedited Transfer of Spent Fuel to the ISFSI

(continued)

- DCPD maintains procedures and conducts extensive training to respond to a potential SFP accident. These mitigative strategies are maintained in accordance with its License Conditions of the DCPD Units 1 & 2 Facility Licenses.
- Conclusion: While PG&E planned to conduct “expedited spent fuel transfer” for decommissioning, this approach was to minimize the overall decommissioning schedule and resulting cost to ratepayers. The expedited spent fuel transfer was not to address a safety concern. Because PG&E, UCLA, and NRC conclusions demonstrate that spent fuel is safely stored in both the SFPs and at the DCPD ISFSI, PG&E currently plans to conduct offload campaigns on a nominal, non-expedited two-year frequency to maintain available SFP space for continued DCPD operations. This frequency will be reevaluated if the State approves legislation to operate DCPD past 2030.
- Recommendation to the DCISC: DCISC should routinely review spent fuel transfer campaign planning and this activity should be added to the recurring Open Items List. (D9/25FF3.4)



Meet with NRC Resident Inspectors

- Items discussed included:
 - The NRC's continued desire to present the results of its annual assessment of DCPD performance at the DCISC's standing June Public Meetings. The NRC Senior Resident stated that having the annual assessment report in conjunction with the DCISC Public Meeting reaches a much broader segment of the public.
 - NRC reorganization and changes being implemented to improve efficiency in the Reactor Inspection Program. Impact to the DCPD station is under development at the regional and headquarter offices. The goal is to streamline inspection activities that have had marginal results, including the biennial Problem Identification and Resolution (PI&R) inspection procedure.
 - Recent NRC inspection findings and concerns.

Conclusion: The meeting with the NRC Resident Inspectors was beneficial, and the DCISC should continue the meetings.



Meet with NRC Resident Inspectors (continued)

Recommendations to the DCISC: The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting and is recommended for closure: *The DCISC should review the workflow of interfacing with the NRC Resident Inspector for a hypothetical scenario when a Notification mentions the need for a Compensatory Action but is later identified that the system was in fact operable and did not require a Compensatory Action (D3/26FF-3.4)*



Meet with DCPD Officer

- The Fact-Finding Team met with Blair Jones, Vice President Business and Technical Services.
- The group discussed the Fact-Finding agenda and other issues of mutual interest.

Conclusion: The regular meetings with DCPD Officers and Directors continued to be beneficial for both organizations.



Flow Accelerated Corrosion (FAC)

- The DCISC FFT met with a Senior Flow Accelerated Corrosion SME, for an update on the health of the stations FAC Program. DCISC last reviewed the FAC Program during its May 2022 Fact-Finding Meeting.
- FAC refers to a phenomenon wherein the combination of the removal of a magnetite (iron corrosion) layer by fluid flow and the subsequent increased corrosion of the exposed metal can lead to rapid wall thinning and leaks or ruptures.
- The objective of the DCPD FAC Program has been to provide a high degree of confidence against the rupture of FAC-susceptible piping systems, and in fact, most FAC-susceptible piping is contained in non-safety related systems.
- A contributing factor in support of DCPD's lower inspection/replacement of FAC piping to date is found in the mixture of alloys for the original steel used to construct key portions of the fluid systems at both plants. More specifically, DCPD Unit 1 was built with a typical mixture of alloys for carbon steel.



Flow Accelerated Corrosion (FAC) (continued)

- DCP Unit 1 steel was created with a mixture of alloys for carbon steel, Chromium was allowed, but not required, and no Chromium was added to Unit 1's steel.
- DCP Unit 2 steel was built of the same contracted alloys but Chromium was included in the steel at .4%, as allowed by the contract.
- As a result, Unit 2 has more resistant steel, resulting in significantly lower amount of piping that needs to be replaced during a refueling outage.
- During refueling outages FAC Engineering also takes measurements of wall thickness and predicts which outage will require a FAC pipe replacement. The engineers are very careful to avoid an operating pipe breach during plant operations, which may cause a unit power reduction or shutdown.



Flow Accelerated Corrosion (FAC) (continued)

- The FAC Program, like many at the station, reduced its analysis of potential pipe FAC issues during the period the Units were to shutdown, at the end of 1R24 & 2R24. This resulted lack of emergent/planned repair or replacement for outages 1R23 through 2R24.
- The FAC Program has recovered well in increasing staff with the license renewal activities behind it for the next few years, perhaps 20 years.
- Refueling Outages 1R25 and Refueling Outage 2R25 had completed the periodic cycle for in-service inspections under the ASME Section XI Code and as such, no code-required inspections needed to be deferred to future outages.
- Conclusion: DCCP continues to maintain a mature and proactive Flow Accelerated Corrosion Program that remains supported by plant management, engineering organizations, and industry operating experience.



Station Update

The DCISC FFT met remotely with Justin Rogers, Station Senior Director, for a Station Update.

- Power Production – Full power at both units. Unit 2 will reduce power to 50% in two weeks to allow for cooling tunnel debris removal. The unit will return to 100% after both tunnels have been.
- Station Performance Indicators – Industrial Safety (IS) has experienced an injury to a worker resulting in a broken arm. As a result, IS has moved off of the Green Status for the station.
- Refueling Outage 1R26 preparations are underway, scheduled to begin on October 14, 2026.



Station Update (continued)

- Major Projects under way include: Upgrade of Control Room/Simulator Annunciators; Nuclear fuel transfer system upgrades; Circulating Water Pump refurbishment; Main Generator Relay Replacements; and Tri Conex System hardware upgrades.

Conclusion: DCPD's update on recent station activities was informative.



Evaluated Emergency Preparedness Exercise

- The purpose of this September 2, 2026 exercise was to develop and maintain the Emergency Response Organization's (ERO's) key skills to protect public health and safety in the unlikely event of a radiological emergency.
- The exercise had as its objective evaluating the following activities:
 - Communications between onsite and offsite Emergency Response Facilities
 - Coordination with offsite response organizations
 - Dissemination of information to the public via media channels
 - Operational and engineering assessment of accident sequences
 - Engineering assessment, repair plan development, and repair of critical equipment under emergency conditions
 - Mitigative action implementation through the simulated repair of equipment
 - Protection of workers (radiological or physical) during emergency response



Evaluated Emergency Preparedness Exercise (continued)

- The Emergency Plan (EP) Exercise was designed to evaluate the proficiency of DCPD personnel in implementing the principal Emergency Plan functions in response to a radiological emergency due to a various actions that results in a radiological exposure.
- The FFT started its observation activities in the site Simulator to witness the initial stages of the Emergency Drill. Once a Site Area Emergency was declared, the FFT then traveled to the Emergency Operation Facility (EOF), located on Kansas Rd, San Luis Obispo. Upon Arrival the EOF building was in the process of staffing the support and recovery management personnel from all functional areas of the plant and county.
- The EOF building housing the EOF also contains the San Luis Obispo County (SLO) Emergency Services, SLO County Police and Fire Departments, California Highway Patrol (CHP), and other County Support personnel.



Evaluated Emergency Preparedness Exercise (continued)

- The FFT observations were largely focused on the EOF, which had the technical staff to assess and declare Emergency Classifications, Classification upgrades, and associated Protective Action Recommendations (PAR) needed to protect the health and safety of individuals located in affected Precautionary Action Zones (PAZ).
- The FFT also observed activities in the Joint Information Center, which consisted of a PG&E Spokesperson, a PG&E Public Information Officer, an SLO County Spokesperson, and the SLO County Certified Public Health Physicist. The spokespersons held periodic mock news conferences with mock media asking questions. There were also numerous written news releases in both English & Spanish by PG&E, SLO County, and National TV correspondents (Note: No actual news releases occurred.)
- Overall, the Emergency Drill appeared to function at a high level of performance and understanding the responsibilities of each group in the Emergency Response Team. EOF personnel used the “update” call to get the attention of all personnel, and in those updates, there was an expectation that the EOF members would quickly point out any shortcomings the EOF Team needed to focus on. The UDAC Team operated in a professional manner, the radiological data was being analyzed and recommendations were discussed with each of the other teams.

(continued)



Evaluated Emergency Preparedness Exercise (continued)

Conclusions: The September 2, 2026, DCPD Evaluated Emergency Preparedness (EP) Exercise appeared well designed and implemented. In this EP Exercise scenario, the activities simulated required a quick change in plant status, going from two units at full power to a Site General Emergency. That required the Control Room to skip over the Unusual Event and Alert conditions. The scenario also involved a sufficiently large release to require declaration of a General Emergency. The Nuclear Regulatory Commission (NRC) evaluation of the EP is not yet available at the time of this writing, and the DCISC will review its evaluation when released. The DCISC should review the DCPD drill critique and NRC evaluation of PG&E and Federal Emergency Management Agency (FEMA) evaluations of off-site emergency support agencies, in a future fact-finding meeting when these reports are completed and available to DCISC.

Recommendations to the DCISC: The DCISC should review the DCPD September 2, 2026 Evaluated Emergency Exercise drill critique and NRC evaluation in a future fact-finding meeting when they are completed. (D9/26FF-3.9)



Emergency Preparedness Meetings

- DCISC has met with San Luis Obispo (SLO) County OES, Cal OES, and California Men's Colony to discuss emergency preparedness.
- The DCCP Emergency Plan and the Emergency Prepared Calendar described 12 Protective Action Zones (PAZ), and 3 Public Education Zones (PEZ). The State expanded the EPZ's so that it is much larger than the EPZ defined by the NRC.
- The expanded EPZ defined by the State is divided into twelve smaller Projective Action Zones. PAZ's 1 through 5 correspond to the ten-mile area where FEMA has primary oversight responsibilities for emergency planning. The State has primary oversight responsibilities for PAZ's 6 through 12, outside of the ten-mile area.
- Residents in PAZs 1 through 12 may be asked to take protective actions, such as take shelter in place or evacuate if there is a major emergency.
- "The State will assist the County in adequate plan review and coordination of PAZs 6 through 12. At the request of the County, if additional support is necessary, courtesy review would be requested of FEMA..."



Emergency Preparedness Meetings (continued)

- “Distribution of informational materials directed at transient populations are located in places likely to be frequented by transients, such as gas stations, motels/hotels, parks and recreation areas, marinas, shopping malls, major employers, etc. Plans/procedures include a list of the locations where such materials are posted, as well as a mechanism for annual updates.”
- DCSIC sampling of hotels, motels, stores, commercial businesses, etc. appear not to be aware of the what to do if the Sirens are activated.

Conclusion: : The DCISC found that for Protective Action Zones (PAZs) 1–12, Emergency Plans, census, sirens, public communication goals and evacuation time estimates and for Public Education Zones (PEZs) 13–15, public communication goals are described in the Emergency Plan updates that PG&E periodically submits to the NRC. Implementation of Emergency Preparedness in PAZs 6–12 is done by San Luis Obispo (SLO) County, with support from the California Office of Emergency Services, and (at county’ s request) review and advice from the Federal Emergency Management Agency. Implementation of public education in PEZs 13–15 is done by the county.



Emergency Preparedness Meetings (continued)

Recommendation to the DCISC:

The DCISC should follow-up with SLO County to identify the effectiveness with which public communication goals are met, as described in the DCPP Emergency Plan, license document, and Federal Emergency Management Agency Program Manual for Radiological Emergency Preparedness Planning, Standard G for Public Communication, including in PEZs 13–15 and in PAZs 6–12, including public communication to transient populations, people with access and functional needs, limited English proficiency communities, parents/guardians, agricultural communities, people for whom evacuation may be difficult, and facilities housing incarcerated populations. (D9/26FF–3.10)



Offsite Emergency Protective Action Measures

The FFT visited California Men's Colony (CMC) on September 3rd, 2026 and met with Mario Fernandez; Captain, and others that have had discussions with San Luis Obispo County OES and CMC exploring the topic of sheltering in place as a protective measure in case of an Emergency.

- The purpose of the meeting is to better understand the interface between the Diablo Canyon Nuclear Power Plant (DCPP) emergency preparedness and the emergency plans for California Men's Colony, which is located in DCPP's Emergency Preparedness Zone 8 (EPZ 8).
- Unlike the rest of the EPZs, which have either sheltered in place or evacuation as possible Protective Action Measures, correctional facilities only shelter in place.
- Several Topics were discussed including: CMC Emergency Plan and Staff Training, Sheltering in place options, Support needs for radiological incidents, and available communication channels.
- The FFT inquired about communications to families of inmates during a sheltering in place event, concerning the well-being of the inmates. A follow-up conversation with the Inmate Family Council was proposed.

(continued) 30



Offsite Emergency Protective Action Measures (continued)

Conclusion: An emergency operational plan is in place for responding to Radiological Incidents at CMC. CMC welcomes further integration into county radiological emergency preparedness drills and training. Interfacing with the Inmate Family Council may be productive in addressing some short comings.

Recommendations to DCISC:

- The DCISC should follow-up with CMC regarding a meeting with building facilities who can describe the ventilation options and corresponding time-scales during a sheltering in place event. (D9/26FF-3.11A)
- The DCISC should follow-up with CMC with a meeting of the Inmate Family Council to learn about public communication and management of radiological emergencies at CMC. (D9/26FF-3.11B)



Offsite Emergency Protective Action Measures (continued)

Recommendations to DCISC: (cont' d)

- The DCISC should follow-up with SLO County to understand procedures for sheltering in place, including ventilation guidelines and corresponding timelines for sheltering in place implementation for large buildings, and to evaluate appropriateness of plans and communication of plans for KI distribution and usage. (D9/26FF-3.11C)
- The DCISC should follow-up with SLO County on the integration of emergency preparedness, training, and drills with state and county correctional facilities, all are located in PAZ 8, and all of which have the unique feature of sheltering in place as the designated protective action measure. (D9/26FF-3.11D)



Public Outreach–Interview with Local Radio Station

- On August 18, 2026, Dr. Scarlet conducted an interview with local radio host, Kendra Hanna, from KCBX FM 90–1 located in the city of San Luis Obispo, Ca. Dr. Scarlet provided a quick bio of her education and role in the DCISC. Dr. Scarlet stated that she was an Independent Neutral Academic and generally listens to the public to get their point of view. When having a dialog with a skeptical member of the public, she speaks with a neutral voice to keep the communications flowing.
- Dr. Scarlet's main theme was to reach out and engage with public messengers, like Ms. Hanna, to keep a level view of safe, reliable, and trustworthy power plant operations. She encouraged the sharing of her interview as a means to reach out to others that may have questions regarding the Diablo Canyon Power Plant.
- Conclusion: Taking the time to reach out to an interested public stakeholder is a very important part of gaining the trust of the organization they represent. Being able to participate in a radio broadcast allows for greater reach to the public in general, and specifically to those who choose to listen, learn, and get engaged.



Public Outreach-Interview with Local Radio Station

- Recommendations to DCISC: Establish a plan and allocate the necessary resources for DCISC's public engagement scope. (D9/26FF-3.12)



Questions?



Dr. Raluca Scarlat, Member, with Richard St. Onge, Consultant

1. Reactor Vessel Embrittlement Sample Capsule B Report
2. Independent Spent Fuel Storage Installation (ISFSI) Spent Fuel Canister Inspections
3. Westinghouse Issuance of 10 CFR Part 21 Reporting of Defects and Noncompliance
4. Spent Fuel Pool Arrangement and Expedited Transfer of Spent Fuel to the ISFSI
5. Meet with Nuclear Regulatory Commission (NRC) Resident Inspector
6. Meet with DCCP Officer
7. Flow Accelerated Corrosion
8. Station Update
9. 2026 Evaluated Emergency Preparedness Drill
10. Emergency Preparedness Meetings
11. Offsite Emergency Protective Action Measures
12. Public Outreach-Interview with Local Radio Station