

DIABLO CANYON INDEPENDENT SAFETY COMMITTEE

Report on

Fact-Finding Meetings at DCP

June 29, August 18 and 19, September 1 and 2, 2026

by

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with contributions from

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1.0 SUMMARY

The results of the DCISC June 29, 2026, August 18 and 19, 2026, and September 1 and 2, 2026, Fact-Finding Meeting at the Diablo Canyon Power Plant (DCPP) in Avila Beach, CA, are presented. The subjects addressed and summarized in Section 3 are as follows:

1. Reactor Vessel Embrittlement Sample Capsule B Report
2. Independent Spent Fuel Storage Installation (ISFSI) Spent Fuel Canister Inspections
3. Westinghouse 10 CFR Part 21 Report 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 DCP Officer
7. Flow Accelerated Corrosion Program
8. Station Update
9. Evaluated Emergency Preparedness Exercise
10. Emergency Preparedness Meetings
11. Offsite Emergency Protective Action Measures
12. Public Outreach – Interview with Local Radio Station

2.0 INTRODUCTION

This Fact-Finding Meeting at the DCP was held to evaluate specific safety matters for the DCISC. The objective of the evaluation was to determine if Pacific Gas and Electric's (PG&E's) performance is appropriate and whether any areas revealed observations which are important enough to warrant further review, follow-up, or presentation at a public meeting. These safety matters include follow-up and/or continuing review efforts by the Committee, as well as those identified as a result of reviews of various safety-related documents.

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Section 4 – Conclusions, highlights the conclusions of the Fact-Finding Team (FFT) based on items reported in Section 3 – Discussion. These highlights also include the FFT’s suggested follow-up items for the DCISC, such as scheduling future Fact-Finding Meetings on the topic, presentations at future public meetings, and requests for future updates or information from DCPD on specific areas of interest, etc.

Section 5 – Recommendations, presents specific recommendations to the DCISC and PG&E proposed by the FFT. These recommendations will be considered by the DCISC. After review and approval by the DCISC, this Fact-Finding Report, including its recommendations, will be provided to PG&E. The Fact-Finding Report will also appear in the DCISC Annual Report.

3.0 DISCUSSION

3.1 Reactor Vessel Embrittlement Sample Capsule B Report

On June 29, 2026, the DCISC FFT received a briefing from PG&E on the report generated from testing and evaluation of Unit 1 Reactor Vessel Embrittlement Sample Capsule B (the “Capsule B Report,” Reference 6.1) which was removed from the vessel on April 21, 2025. A follow-up briefing was held on July 1, 2026, in response to additional questions from Dr. Scarlet. In addition to the FFT, the following attended the briefings:

- DCISC Member Dr. Raluca Scarlet
- DCISC Counsel Robert Rathie.
- DCISC Consultants: Richard McWorter, Richard St Onge, Lisa Darden, Dr. Andy Kadak, Mark Kirk
- PG&E staff members Blair Jones, Vice President Business and Technical Services; Tom Jones, Regulatory, Environmental and Repurposing Senior Director; Saul Gomez, Government Relations Senior Director; Philippe Soenen, Strategic Initiatives, Performance Improvement and Organizational Effectiveness Director; Josh Harmon, State Agency Relations Principal; and Michelle Olsofsky, License Renewal Engineer.
- Westinghouse staff members included Michael Frances, Principal Project Manager; Brett Lynch, Principal Vessel Integrity Engineer; Karina Assis, Principal Materials Engineering; Margret Long, Senior Mechanical Engineer; and Andy Hawk, Lead Engineer Radiation Engineering and Analysis.
- The June 29, 2026, briefing was attended by Karina Assis, Michael Frances, Margaret Long, Andrew Hawk, and Brett Lynch. The July 1, 2026, briefing was attended by

Karina Assis, Brett Lynch, Margaret Long, and Andrew Hawk. The Capsule B WCAP report identifies Margaret Long and Marcos Alberto Guillen as authors and Donald Lynch and Andrew E. Hawk as reviewers of the report.

In the DCISC's 34th Annual Report (Reference 6.2), the following recommendation was opened:

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)

PG&E met the NRC-required timeline of report submission to the NRC no longer than 18 months after capsule withdrawal (Oct. 21, 2026). PG&E kept DCISC apprised of the evolving timeline for submission of the Capsule B report, to allow for scheduling a debrief meeting as soon as the report was available. While initial timeline projections for analysis of Capsule B results were predicting May as a time-frame, the data collection, associated data reduction, curve fitting, and technical evaluation activities continued into late June 2026. Internal report metadata indicates that the Charpy curve-fitting analyses used to establish transition temperature shifts and upper-shelf energy changes were generated between May 28 and June 10, 2026. Specifically, CVGRAPH analyses for the correlation monitor material were produced on May 28 and May 29, 2026; surrogate weld analyses were generated on June 7, 2026; surveillance weld analyses on June 9, 2026; and lower shell plate analyses on June 10, 2026. The Westinghouse report was finalized and approved on Friday, June 26, 2026, and PG&E transmitted the report to the NRC under Letter DCL-26-046 dated Monday, June 29, 2026 (Reference 6.3).

The first confidential briefing to a Member of the DCISC occurred the morning of Monday, June 29th, 2026, shortly before submission of the report to the NRC that same day. All of the FFT requests for additional information and additional briefings were met by PG&E and Westinghouse without undue delay.

The Capsule B Westinghouse and PG&E report concluded that Unit 1 meets the reactor vessel integrity criteria for at least 60 years of operation (through November 2, 2044). The results were identified as consistent with the conclusions contained in the NRC's June 2025 License Renewal Safety Evaluation Report (Reference 6.4) and the existing Reactor Vessel integrity analysis of record (Reference 6.5).

The NRC staff's assessment of the Capsule B report was issued on Sept. 24th, 2026, in a letter from the US NRC Materials Branch Senior Materials Engineer On Yee issued to Branch Chief Office of Nuclear Reactor Regulation (NRR) Angela Buford. The NRC staff assessment included a review of the ways in which the methodology for neutron fluence evaluation, temperature of irradiation, credibility analysis, transition temperature shift, and upper shelf energy met the regulatory requirements. The NRC staff's assessment concluded:

- that "the report satisfies the requirements of 10CFR50 Appendix H" and
- that PG&E's "assessment of test results were appropriate and based on Regulatory Guide 1.99, Revision 2.", and

- “that a revision to the licensee’s currently Pressure and Temperature Limits Report (Revision 17) is not warranted.”

A copy of PG&E’s presentation materials for this briefing is included as Enclosure 1 to this report and also made available as a stand-alone document on the DCISC website. PG&E staff concluded that the results of the Capsule B Report confirmed that Unit 1 continues to meet NRC requirements for Reactor Vessel integrity and remains safe for at least 60 years of operation. The Capsule B results were identified as consistent with the conclusions contained in the NRC’s June 2025 License Renewal Safety Evaluation Report (Reference 6.4) and the existing Reactor Vessel integrity analysis of record (Reference 6.5). PG&E and Westinghouse provided follow-up responses and supporting documentation addressing questions raised by a DCISC Member during the debrief meetings on June 29th and July 1st. In the two briefings with PG&E and Westinghouse, the following topics were discussed:

- Numbers and composition of material samples tested and the test results
- Dosimetry (fluence) and thermal data for the sample capsule
- Data credibility and criteria for data credibility (as defined by NRC regulations)
- Quality assurance/quality control (QA/QC) and metrology for the temperature, dose, and mechanical property testing
- Independent review of data post processing and interpretation

Background on embrittlement

Neutron irradiation causes aging (in the form of embrittlement) of the reactor vessel. The magnitude of irradiation dose is quantified by fluence of fast neutrons that the material has experienced at a particular plane in the material (n/cm²). As for all ferritic steels, the reactor pressure vessel steel experiences a transition from brittle behavior at low temperatures to ductile behavior at high temperatures. The temperature at which the steel transitions from brittle to ductile behavior increases with increased irradiation dose. The "limiting material" in a reactor pressure vessel is the steel that has a transition temperature closest to the NRC's pressurized thermal shock screening (PTS) criterion found in 10 CFR 50.61. The limiting material for Unit 1 is the longitudinal weld 3-442C, which was last ultrasonically-inspected following the requirements of the ASME Code in 2025. The thickness of the reactor pressure vessel is more than 8 inches at the beltline. By the end of 2026, 41 years after the start of Unit 1 operation, the inner wall of this weld will have experienced approximately 1.5×10^{19} n/cm² fast neutron fluence. This fluence represents approximately 36 effective full power years (EFPY) of Unit 1 operation. Neutron irradiation scales with power level and hence energy produced; since reactors sometimes operate at reduced power and have outage periods for refueling and maintenance, the EFPY is less than the age of the reactor. For Unit 1, the age of the reactor is referred to as the years that have passed since 1985, irrespective of EFPY of operation. EFPY, and hence fluence, will depend on the power level and duration of outages of the specific unit.

For convenience, Table 1 below correlates the values of fluence with the corresponding operational calendar year, age of reactor, and effective full-power years (EFPY) when these fluence levels have been or could be reached for the Unit 1 RPV. Values are derived from Tables 6-6, 7-1, and

5-10 in Capsule B report (ML26180A296). After 2045, hypothetical projections are made assuming 1 EFPY per year and $(0.04)10^{19}$ n/cm² per EFPY.

Table 1. Correlation of fluence with the corresponding operational calendar year, age of reactor, and effective full-power years (EFPY) when these fluence levels have been or could be reached for the Unit 1 RPV. Values are derived from Tables 6-6, 7-1, and 5-10 in Capsule B report (ML26180A296); after 2045, hypothetical projections are made assuming 1 EFPY per year and $(0.04)10^{19}$ n/cm² per EFPY.

Fluence (10^{19} n/cm ²)	Approximate Year Reactor Reaches this Fluence	Corresp. Reactor Age (years)	Corresp. EFPY	Irradiated samples with corresponding fluence
0.293	1993	8	5	Capsule S
1.07	2015	30	26	Capsule Y
1.24	2020	35	31	Capsules C&D (removed from reactor, not analyzed)
1.35	2024	39	34	Capsule V
1.42	2026	41	36	
1.6	2030	45	41	(fluence and EFPY estimated by interpolation)
2.14	2045	60	54	(values per Amendment 1 to the PG&E license renewal application, ML24289A118)
3	2066	81	75	(hypothetical projection)
3.62	2083	98	92	Capsule B (surrogate weld)
3.91	2090	105	99	Capsule B
6	2143	158	152	(hypothetical projection)
Note: The year, reactor age and EFPY are not to be misunderstood as characterizing the conditions of capsule removal. The year, reactor age, and EFPY in this table are specifically the conditions at which the respective fast neutron fluence would be reached on the inner face of the reactor pressure vessel (RPV). The maximum of across all axial and azimuthal positions on the RPV is reported.				

Unit 1 Reactor Vessel Inspections (2025 ISI)

To support review of the 2025 Unit 1 reactor vessel in service inspection (ISI) report, PG&E provided the drawing "ISI RPV Unit 1 Weld ID", which identifies the locations and designations of the reactor vessel welds. According to PG&E, all Unit 1 reactor vessel welds were inspected during the 2025 inspection campaign.

Questions were raised regarding reactor vessel weld heat 27204, which is associated with the limiting reactor vessel embrittlement material. PG&E and Westinghouse reported that the following beltline welds were fabricated using weld heat 27204:

- Intermediate Shell Longitudinal Weld Seams 2-442A, 2-442B, and 2-442C.
- Lower Shell Longitudinal Weld Seams 3-442A, 3-442B, and 3-442C.

Westinghouse further clarified that, from a neutron fluence perspective, Lower Shell Longitudinal Weld 3-442C is the limiting beltline weld for Unit 1. That weld, along with the remaining heat 27204 welds listed above, was ultrasonically examined during the 2025 inspection. The FFT reviewed the inspection report and confirms that the inspection did not identify problematic flaws and met the criteria for the % inspected.

The FFT noted that the ultrasonic examination report previously reviewed appeared to focus primarily on reactor vessel penetrations and questioned whether the limiting beltline welds had

in fact been examined. In response, PG&E provided selected pages from the 2025 reactor vessel examination report documenting ultrasonic examinations of welds 2-442A through 2-442C and welds 3-442A through 3-442C.

Previous Unit 1 Reactor Vessel Inspections

The FFT raised with PG&E questions concerning the timing and scope of previous Unit 1 reactor vessel volumetric examinations. Westinghouse clarified that the last full reactor vessel examination was performed in 2005. The subsequent examination conducted in February 2014 was a partial reactor vessel examination. Welds 2-442A, 2-442B, and 2-442C were examined during the 2014 inspection, while welds 3-442A, 3-442B, and 3-442C were among the welds not examined. To support review of historical inspection activities, PG&E provided excerpts from both the 2005 and 2014 reactor vessel examination reports focusing on the beltline welds discussed above.

Extension of Reactor Vessel Inspection Interval

The FFT requested information regarding the basis for extending the Unit 1 reactor vessel inspection interval from 10 years to 20 years. PG&E reported that it submitted the extension request to the NRC in 2014 under correspondence DCL-14-074. Supporting correspondence included PG&E's response to NRC requests for additional information (DCL-15-040) and the subsequent NRC approval package ("150619, Request RPV-U1-Extension"). These documents were provided to the FFT.

Capsule B Thermal Monitors

The FFT requested photographs of the thermal monitors recovered from Capsule B together with examples of melted thermal monitors for comparison. Westinghouse provided three photographs of the Capsule B thermal monitors (files BOT 2.tif, MID 2.tif, and TOP 2.tif). The photographs were obtained using the camera equipment available within the Westinghouse hot cell. Westinghouse reported that an experienced technician visually examined the monitors and determined that none of the thermal monitors had melted. Westinghouse was unable to provide high-resolution photographs of melted thermal monitor wires for comparison because such photographs were not available.

Charpy Testing and Temperature Control

Questions were raised regarding calibration accuracy and measurement uncertainty associated with the Charpy impact testing and temperature measurement systems. For the Charpy impact test machine:

- Verification range extends from 6.3 ft-lb to 211 ft-lb, representing approximately 80 percent of machine capacity.
- The largest observed deviation from National Institute of Standards and Technology (NIST) reference specimens was 3.2 percent.

Temperature control was provided through calibrated temperature baths, each utilizing a dedicated thermocouple. The temperature controller accuracy was reported as $\pm 1^{\circ}\text{F}$.

Calibration History

Westinghouse reported that annual NIST verification of the Charpy testing machine was last completed on July 10, 2025.

Thermocouple calibrations were completed July 10, 2025, for testing of non-reconstituted specimens, and November 18, 2025, for testing of reconstituted specimens. Thermocouples are calibrated on a six-month frequency.

Dosimetry Standards

The FFT requested identification of the standards used for reactor dosimetry counting and analysis. Westinghouse stated that its internal procedures governing high-purity germanium detector calibration, dosimetry sample preparation, and sample assay were developed in accordance with several ASTM reactor dosimetry standards, including:

- ASTM E3376, Standard Practice for Calibration and Usage of Germanium Detectors in Radiation Metrology for Reactor Dosimetry
- ASTM E263, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron
- ASTM E264, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel
- ASTM E481, Standard Practice for Measuring Neutron Fluence Rates by Radioactivation of Cobalt and Silver
- ASTM E523, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper
- ASTM E526, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium
- ASTM E704, Standard Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238
- ASTM E705, Standard Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237

These standards form the technical basis for the reactor dosimetry measurements used to support fluence evaluations and reactor vessel embrittlement analyses. The seven elements listed in the standards above cover the entire list of elements irradiated in the Capsule B. Dosimetry wires are thus used for the validation of the calculated fast neutron fluences experienced by the capsule. The FFT did not review the applicability of the above-mentioned standards to the specific condition of Capsule B, but notes that these are otherwise the relevant set of standards to apply. The following table is reproduced from the Capsule B report and summarizes the conclusions of

the neutron dosimetry:

Reaction	Reaction Rate (rps/atom)		M/C
	Measured	Calculated	
$^{63}\text{Cu} (n,\alpha) ^{60}\text{Co}$	2.94E-17	2.68E-17	1.10
$^{54}\text{Fe} (n,p) ^{54}\text{Mn}$	2.88E-15	2.85E-15	1.01
$^{58}\text{Ni} (n,p) ^{58}\text{Co}$	4.02E-15	3.91E-15	1.03
$^{238}\text{U}(\text{Cd}) (n,f) ^{137}\text{Cs}$	1.34E-14	1.38E-14	0.97
$^{237}\text{Np}(\text{Cd}) (n,f) ^{137}\text{Cs}$	1.05E-13	1.06E-13	0.99
$^{59}\text{Co} (n,\gamma) ^{60}\text{Co}$	2.03E-12	1.77E-12	1.15
$^{59}\text{Co}(\text{Cd}) (n,\gamma) ^{60}\text{Co}$	8.12E-13	8.31E-13	0.98
Average of M/C Results			1.02
Standard Deviation (%)			4.9

The measured-to-calculated (M/C) reaction rate ratios for the Capsule B threshold (i.e., non Co-59) reactions range from 0.97 to 1.03, and the average M/C ratio is $1.02 \pm 4.9\%$ (1σ). This direct comparison falls within the $\pm 20\%$ criterion specified in Regulatory Guide 1.190. This comparison, along with the other plant-specific surveillance capsule dosimetry comparisons provided in Appendix A, validates the analytical results described in Section 6.2.

Quality Assurance Oversight

Westinghouse has corporate experience in the performance and reporting of surveillance capsule results dating back to the 1970s. Diablo Canyon is a Pressurized Water Reactor, and Westinghouse performs the testing on virtually all capsules from pressurized water reactors in the USA, of which there are 60 operating units designed and supplied by Westinghouse or Combustion Engineering (acquired by Westinghouse).

Quality assurance programs certify the quality of the data and the transparent documentation of the quality and include such considerations as training, expertise, review and audits. Thus, the FFT inquired into the review the metrology and corresponding quality assurance and quality control pertaining to data reported in the Capsule B report.

The FFT asked when the testing processes had last been independently audited to verify compliance with applicable ASTM standards. Westinghouse responded that its Quality Management System is designed to comply with applicable regulatory and industry standards, including NRC 10 CFR 50 Appendix B requirements and ISO 9001 quality management requirements. Westinghouse reported that the most recent ISO Quality Management System audit was conducted in 2025. In addition, the FFT was informed that a Nuclear Procurement Issues Committee (NUPIC) audit is scheduled. NUPIC audits are conducted by representatives from multiple nuclear utilities and are used throughout the industry to assess nuclear suppliers and commercial vendors against applicable quality assurance requirements. The FFT was also informed that review of the Westinghouse Quality Assurance Plan by PG&E remains ongoing.

The FFT was given the opportunity to review the PG&E confidential document that defines the procedures for and acceptance criteria for “Design and Analysis Documents Prepared by External Contractors.” In the case of Capsule B, Westinghouse was the external contractor that performed the measurements, data reduction and analysis of the data, as commissioned by PG&E. The specific technical requirements and quality requirements would be specified in the PG&E-Westinghouse contract. This procedure that was afforded the FFT’s review defines the minimum requirements for PG&E’s acceptance of the quality of the work reported by Westinghouse and generally includes review of assumptions, methodology, standards applied, and regulatory requirements met. Since the FFT had the opportunity to discuss with both Westinghouse and PG&E in the two debrief meetings, we can confirm that there was close communication between PG&E and Westinghouse about the content of the Capsule B report. We cannot confirm the level of technical detail of PG&E’s review of the Westinghouse data pedigree and analysis. All technical questions pertaining to data generation, data reduction and interpretation were handled by Westinghouse staff. Questions pertaining to meeting licensing requirements were responded by PG&E. We can confirm that procedures exist for PG&E to establish acceptance and review criteria for data generated by Westinghouse, and that procedures exist for quality assurance oversight for vendors and suppliers to PG&E.

Regulatory Evolution

The committee has also responded to public inquiries on the topic of Capsule B review, with the correspondence being made publicly available the DCISC website as it was created, and the substance of which is reproduced in this section.

The Westinghouse Capsule B report (ML26180A296) was submitted by PG&E to the NRC to demonstrate PG&E’s regulatory compliance with Appendix H to 10 CFR Part 50. This report, together with its PG&E transmittal letter to the NRC references the specific regulations and standards that govern the Reactor Vessel Surveillance program at Diablo Canyon (e.g. 10 CFR 50, 10 CFR 54). Referencing studies other than what is required and accepted by the current NRC regulation was not necessary as part of reporting surveillance capsule results to the NRC. For this reason, PG&E and Westinghouse did not discuss the possible future evolution of NRC regulations in the Capsule B report.

Regulatory evolution on the topic of embrittlement was discussed by DCISC at the June 2026 Public Meeting; see June 17th, 2026 recording time-stamp 4:02 to 4:35, accessible at: https://slospan.org/meeting/dcisc_20260616/. Among the topics of conversation were the 2023 ASTM paper by the NRC authors Widrevitz and Poehler, the 2019 NRC assessment of RG1.99 rev 2, and the 2026 NRC white paper:

D. S. Widrevitz and J. C. Poehler, “History and Development of U.S. NRC Regulatory Guide 1.99,” in “Radiation Embrittlement Trend Curves and Equations and Their Use for RPV Integrity Evaluations,” ed. W. L. Server, M. Brumovský, and M. Kirk (West Conshohocken, PA: ASTM International, 2023), 398–416, <http://doi.org/10.1520/STP164720220041>. ASTM Symposium on Radiation Embrittlement Trend Curves/Equations and Their Use for RPV Integrity Evaluations held in Prague on April 19-22, 2022.

D. S. Widrevitz and M. Gordon, Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99, NRC Technical Letter Report TLR-RES/DE/CIB-2019-2, U.S. Nuclear Regulatory Commission, July 2019. [ADAMS](https://www.nrc.gov/docs/ML1920/ML19203A089.pdf) Accession No. ML19203A089. <https://www.nrc.gov/docs/ML1920/ML19203A089.pdf>

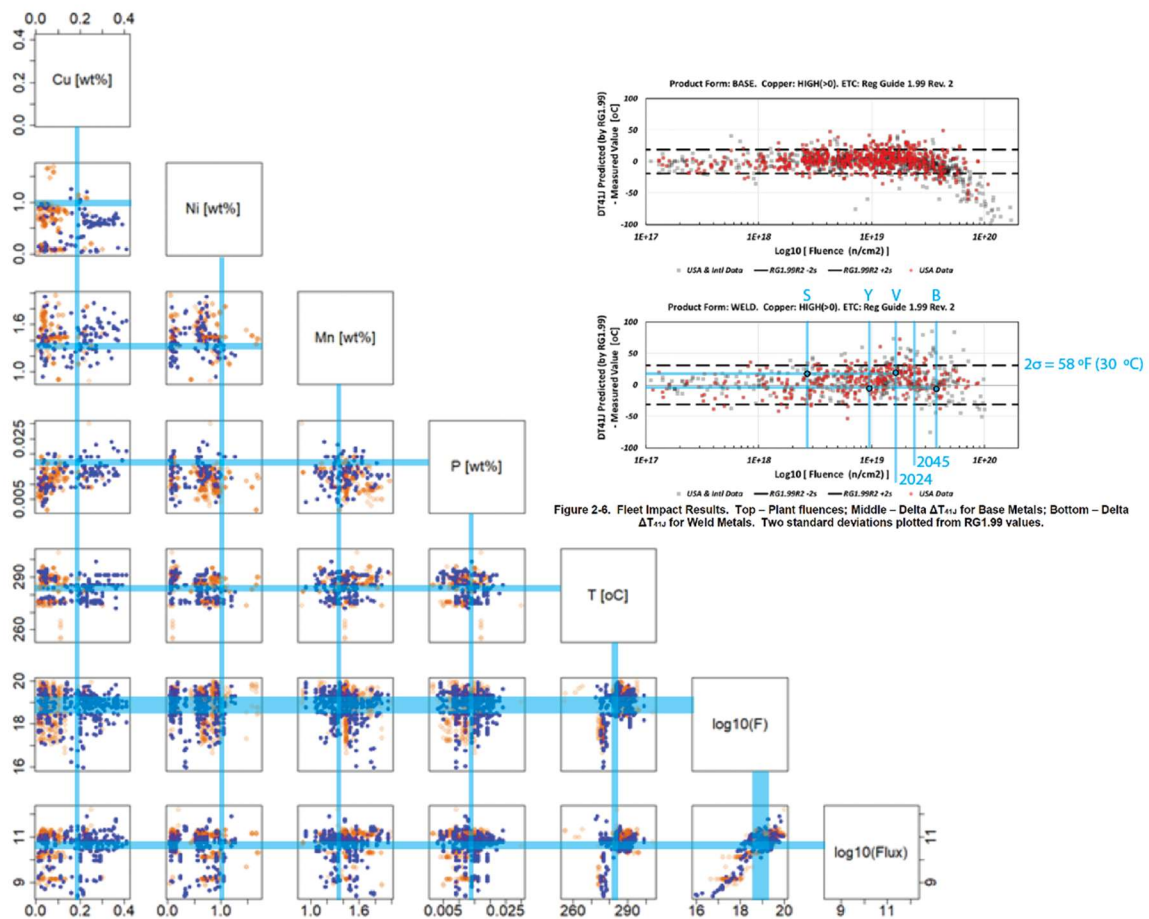
U.S. Nuclear Regulatory Commission, White Paper: Potential Changes to Embrittlement and Surveillance Requirements for High-Fluence Nuclear Power Plants in Long-Term Operation under SRM-SECY-22-0019, June 2, 2026. [ADAMS](https://www.nrc.gov/docs/ML2614/ML26148A433.pdf) Accession No. ML26148A431. <https://www.nrc.gov/docs/ML2614/ML26148A433.pdf>

The ASTM 2023 paper by Widrevitz and Poehler describes the history and development of Regulatory Guide 1.99. This ASTM 2023 paper is part of on-going communication from the NRC concerning its effort to update its embrittlement regulations. Following the 2019 Technical Letter Report, in 2022 the NRC staff requested authorization from the NRC Commissioners to begin rulemaking on this topic (SRM-SECY-22-0019). The staff was authorized by the Commission to begin that work in 2025. In June 2026, the NRC issued a "white paper [that] discusses NRC staff's preliminary technical perspectives developed in response to the direction in SRM-SECY-22-0019 [...] intended solely to describe the staff's initial thoughts and to support stakeholder engagement." As of Oct. 13, 2026, the NRC has issued neither revised guidance nor revised rules.

As an example of the type of regulatory developments that these discuss, it is observed that above a certain fluence, the NRC Regulatory Guide 1.99v2 will begin to under-predict the transition temperature, especially for base metals, with the following points of consideration:

- “of the 177 data points used to generate the RG1.99 trend curve, the highest fluence data points were 8×10^{19} n/cm² ($E > 1$ MeV).the 1.99v2.”
- “the fluence function f of RG1.99 [...]was not chosen to model embrittlement at very high fluence; rather, its performance was judged adequate for fluences below 1×10^{20} n/cm² ($E > 1$ MeV).”
- “The ΔT_{41J} trend curve is stated as being valid between 274°C and 310°C (525°F and 590°F) without correction, although this was later supplemented by suggesting the degree-per-degree adjustment. [...]”
- “As the RG1.99 ΔT_{41J} was created as a curve fit, it is expected that its predictions will be most accurate for inputs within the bounds of the data used to fit the curve.

In the dataset analyzed in the NRC Technical Letter Report, separate treatment is given to base metal and weld metal samples, and 509 PWR and 165 BWR weld data points are included. To illustrate the relevance of this data set to the DCPD U1 limiting weld, Figures 2-3(b) and 2-6 from the 2019 Technical Letter Report is reproduced below with an overlay of gridlines that indicate the values for the surveillance weld samples tested in capsules S, Y, V, and B.



(b) Weld Metals - Blue points are U.S. surveillance data, orange points are international surveillance data.
Figure 2-3. Distribution of variables important to the prediction of embrittlement trends. (a) Base metals and (b) weld metals.

The ASTM 2023 paper also discusses “additional surveillance testing requirements for long-term operation [...] (e.g., beyond 60 years of operation),” in light of larger data scatter observed at fluences above $3\text{--}6 \times 10^{19}$ n/cm². Subsequently, the 2026 NRC White Paper,

"Potential Changes to Embrittlement and Surveillance Requirements for High-Fluence Nuclear Power Plants in Long-Term Operation under SRMSECY-22-0019," Nuclear Regulatory Commission, ML26148A433, June 2, 2026,

proposes a revision of surveillance requirements "to ensure that data [...] with a neutron fluence equal or greater than the peak RPV neutron fluence for extended operation is tested at an appropriate periodicity (e.g., every 20 years);" and it recognizes that "some licensees may have already obtained surveillance data that supports long term operation of the RPV", in Section 3.1.1. Surveillance.

The samples of the limiting material removed with Capsule B had accumulated a fast neutron fluence of 3.91×10^{19} n/cm². The current scope of DCISC does not include considerations of

operations beyond 2030 per California legislature (1.42×10^{19} n/cm² fluence on the inner surface of the Unit 1 vessel) or 2045 per license renewal approved by the NRC (2.14×10^{19} n/cm² fluence on the inner surface of the Unit 1 vessel; Table 1). Since currently available surveillance capsule data for Unit 1 exceeds these value, there would not be a benefit to testing of coupons at even higher fluence values.

Discussion

For aging management of the DCPD “reactor pressure vessels”, there are several tools at the disposal of reactor engineers to inspect, and to manage the aging of the reactor pressure vessel. For example, if exceeded, the NRC's PTS screening criteria prompts additional reactor-specific analysis, or mitigation measures. The adjusted reference temperature (ART) also informs the Pressure - Temperature Limits, as described in Generic Letter 88-11 and cited by SRM-SECY-22-0019:

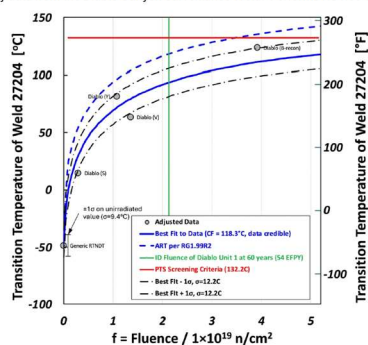
Generic Letter 88-11, “NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations,” issued July 1988,
<https://www.nrc.gov/documents-reports/generic-communications/generic-letters/1988/gl88011>,

“Licensees and permittees should use the methods described in Revision 2 to Regulatory Guide 1.99 to predict the effect of neutron radiation on reactor vessel materials as required by Paragraph V.A. of 10 CFR Part 50 Appendix G, unless they can justify the use of different methods. The use of the Revision 2 methodology may result in a modification of the pressure-temperature limits contained in Technical Specifications in order to continue to satisfy the requirements of Sec. V of 10 CFR Part 50, Appendix G”.

The NRC, in its review of Capsule B results, identified that license renewal analysis for reactor pressure vessel integrity remains unchanged, including the time-limited analysis, and including pressure-temperature limits.

Focusing specifically on the PTS screening criterion of not exceeding 270 °F transition temperature, the outcomes of the analysis are sensitive to the outcome of the credibility analysis. This is because for a credible plant-specific data set, the margin applied to the embrittlement curve is halved (from 28 °F to 14 °F). In the case of using the generic RG1.99 curve margin ($\sigma_A=28$ °F), the PTS screening criterion is reached in the vicinity of $(1.8) \times 10^{19}$ n/cm², which would occur between years 2035 to 2040 (See Table 1 above for fluence to year correspondence. See figures below for Case 1 and Case 2 hypothetical analysis; calculations performed by Mark Kirk, July 22nd, 2026.)

Results from Analysis Case 1
Uses only data from the Diablo Canyon Surveillance Weld. Performs RG1.99-R2 analysis.



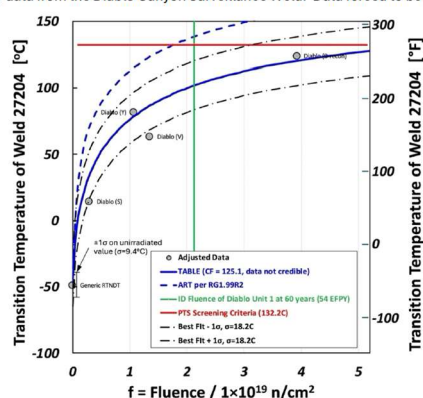
Credibility Assessment	Data are credible Capsule V is outside of $\pm 1\sigma$, $\sigma = 15.6^\circ\text{C}$ for welds 25% of data outside is acceptable
Chemistry Factor (best fit to data)	120.3 °C 213.0 °F
RT _{TS} at 60-year fluence of $2.13 \times 10^{19} \text{ n/cm}^2$ (electrode)	118.2 °C; 14 °C below the PTS screening criteria 244.8 °F; 25 °F below the PTS screening criteria
Fluence at which ART exceeds the PTS screening criteria and corresponding approximate years of service	$3.4 \times 10^{19} \text{ n/cm}^2$ Approximately 95 years
Margin = $2\sqrt{\sigma_0^2 + \sigma_s^2}$	$\sigma_0 = 9.4^\circ\text{C}$ or 17°F , which is as specified in [10CFR50.61] when a generic RT _{TSR} is used, see [Vagins 1982] for the basis. $\sigma_s = 7.8^\circ\text{C}$ or 14.0°F , which is as specified in [RG1.99R2] and in [10CFR50.61], see [Randall 1986] for the basis. Total σ (value of the square-root) = 12.2°C or 22.0°F
Fluence function	$f^{(0.28-0.11\log(f))}$ f is neutron fluence in n/cm^2 divided by $1\text{E}19$

² Fluence from DCL-24-092, Amendment 1 to 60-year LRA, see page 340, NRC ADAMS ML24289A118. Value is for the lower shell longitudinal weld designated as 3-442C.

³ Memorandum to S. Hanauer from M. Vagins, "Findings of the *ad hoc* working group on the selection of RT_{TSR} values," 30 August 1982, United States Nuclear Regulatory Commission, Legacy ADAMS Number 8509200138.

⁴ <https://doi.org/10.1520/S1P230345>

Results from Analysis Case 2
Uses only data from the Diablo Canyon Surveillance Weld. Data forced to be not credible



Credibility Assessment	Forced to be "not credible"
Chemistry Factor (not fit, this value is from RG1.99 Tables)	125.1 °C 225.1 °F
RT _{TS} at 60-year fluence of $2.13 \times 10^{19} \text{ n/cm}^2$	138.3 °C; 6 °C above the PTS screening criteria 281.0 °F; 11 °F above the PTS screening criteria
Fluence at which ART exceeds the PTS screening criteria and corresponding approximate years of service	$1.8 \times 10^{19} \text{ n/cm}^2$ Approximately 51 years
Margin = $2\sqrt{\sigma_0^2 + \sigma_s^2}$	$\sigma_0 = 9.4^\circ\text{C}$ or 17°F , which is as specified in [10CFR50.61] when a generic RT _{TSR} is used, see [Vagins 1982] for the basis. $\sigma_s = 15.6^\circ\text{C}$ or 28.0°F , which is as specified in [RG1.99R2] and in [10CFR50.61], see [Randall 1986] for the basis. Total σ (value of the square-root) = 18.2°C or 32.8°F
Fluence function	$f^{(0.28-0.11\log(f))}$ f is neutron fluence in n/cm^2 divided by $1\text{E}19$

In the latter case, with or without Capsule B data, with or without Palisades data and surrogate weld date, the PTS screening criterion is not reached until after 60 years of operation (year 2045). Since the dataset of Unit 1-specific embrittlement data already contains three plant-specific points and two sister-plant data points and meets already the credibility requirements for this subset of data, additional data points that fall outside of 1σ data points would not make the dataset not credible, because only 68% of the dataset needs to be within 1σ of the fit, and prior to Capsule B data this criterion is already met by 100% of the points. Capsule B data points both fall just outside of 1σ , and this brings 71% of the data set to within 1σ , still above 68%. Notably Capsule B data points both fall within 2σ of the fit. The above discussion could be repeated with a study of the sensitivity of the credibility analysis to the degree-per-degree adjustment.

Thus, in what concerns current regulation for PTS screening, the primary question to answer with confidence is about the classification of the Unit 1 embrittlement data set as credible or not credible.

Lastly, in order to meet the license extension requirement of having embrittlement measurements at a fluence higher than the fluence at the end of the licensing period, it must be deemed that the Capsule B results are valid results. NRC confirmed that the correct methodologies were employed, and the quality assurance programs ensure that the methodologies were correctly applied. This FFT did not have the opportunity to verify the detailed applicability of the quality assurance program to all of the Capsule B activities, but questions were asked of PG&E and Westinghouse on this topic, and thus a sampling of topics of relevance in this area is provided in this fact-finding report. All of the sampling performed thus far demonstrates confidence in the data and competence and expertise of Westinghouse and PG&E in correctly performing and correctly interpreting the measurements.

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.

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 Unit 1 embrittlement, including verification and certification processes, quality assurance and quality control.
2. The type of decisions that are informed by capsule embrittlement data to improve confidence in plant safety, including but not limited to: the roles 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)

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)

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

The FFT met with the primary personnel involved in the ISFSI Canister Inspections while directly observing canister inspections on the ISFSI pad on August 18 and 19, 2026. The personnel included Eric Brackeen, ISFSI Programs Manager; Kyle Duke, ISFSI Program Supervisor; Michelle Olsofsky, Senior ISFSI Engineer; Saya Day, ISFSI Engineer; Terry Carraher, Inservice Inspections (ISI) Senior Engineer, Certifications in VT-I, and VT-III Visual Inspections; and Micheal Wagner, ISFSI Program Assistant. The Team included contract personnel from Holtec, Integrity Assessment Group (IAG), and Robotic Technologies of Tennessee (RTT). Several of the contract personnel held American Society of Mechanical Engineers (ASME) visual inspection qualifications. The US Nuclear Regulatory Commission also had two inspectors t who had access to the inspection activities.

The ISFSI Canister Inspection Campaign kickoff started on August 18th, 2026 and was witnessed for two days by the DCISC Fact-Finding Team. Later during its September 1 and 2 meetings, the DCISC FFT requested an update on the completion of the ISFSI Canister Inspection project, and an update was provided by Eric Brackeen, ISFSI Manager; Kyle Duke, ISFSI Supervisor; Brian Ketletsen, Director of Business & Technical Services; and Carina Corral, Strategic Initiatives Manager.

The DCISC last reviewed this topic during its March 18 and 19, 2026, Fact-Finding Meeting (Reference 6.6), when it provided the following recommendation:

The Independent Spent Fuel Storage Installation (ISFSI) inspection team provided DCISC information on the details regarding the procedures, equipment, training, and contract for the upcoming August 2026 inspection campaign. The DCISC concluded that the readiness of the ISFSI Inspection Team was on track.

The DCISC has been tracking the DCPD plans for inspecting spent fuel stainless-steel canisters for Aging Management Program (AMP) purposes. As part of this planning, there has been concerns about the possibility of Stress Corrosion Cracking (SCC) occurring in the Holtec Canisters being stored on the DCPD ISFSI. DCPD developed the first draft of responses to concerns by a member of the public at the DCISC July 2025, “ISFSU Update”, Fact-finding meeting (Reference 6.7). At the September 2025 FFM, DCPD provided responses to questions and concerns received by the DCISC from a member of the public regarding methods used to inspect Spent Fuel Multi-Purpose Canisters for corrosion. Additional information was requested during this Fact-Finding Meeting and later provided by PG&E (Reference 6.8)

In the aggregate, there were a total of five (5) DCISC 2025-2026 Fact-finding Meetings that were focused on the importance of having the best Industry support and State-of-the-Art Inspection Equipment for the August 2026 ISFSI Canister Inspection campaign. This industry support and attention, drove the development of a crawler, that alongside the high-resolution camera, also had the means to scrub deposits from the canisters without damaging the canister surface.

Background on ISFSI Inspections

As a condition of the Diablo Canyon NRC-issued license to store spent fuel on-site, the Station was required to periodically perform inspections of the canisters present on the ISFSI concrete pad. Diablo Canyon selected eight canisters with the Overpack shielding for AMP purposes, to be inspected every five years. Diablo Canyon selected eight canisters to be inspected as these canisters represent slightly different metal alloys and weld materials that were placed in use early on the ISFSI pad. Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) renewal license letter (License No. SNM-2511) is found in ML24310A118.

Each of the stainless-steel Multi-Purpose Canisters (MPCs) contain 32 spent fuel assemblies that are loaded in the spent fuel pool. During the canister loading process the canister is loaded with spent fuel, capped with a welded 9.5-inch-thick stainless steel shielding lid, drained of spent fuel pool liquid, filled with an inert gas (helium), and permanently enclosed using a welded 0.375-inch-thick closure ring. Canister walls are 0.5-inch thick and the canister baseplate is 2.5 inches thick. Each canister's acceptable peak heat load is approximately 27 kilowatts, with the actual heat load loading varying based on the combinations of the burnup of spent fuel assemblies loaded into each canister. This data has been gathered from the ISFSI FSAR drawings and descriptions.

The spent fuel resides inside the stainless-steel MPC canister, which has been drained, inerted, and welded shut. There are two types of welds being inspected: those that are factory-made by Holtec during manufacturing of the MPC (circumferential and axial) and the two circumferential closure-ring welds made at the site after filling the MPC. The canister is inserted into the overpack which is approximately 20-foot high by 11-foot diameter. The overpack has an open space in its middle that is slightly larger than the canister diameter. The overpack is a steel lined, concrete filled vessel with sides that are approximately 29.5 inches thick, the overpack is comprised of a 1-inch-thick steel outer shell, 27.5-inch-thick concrete, and 1-inch-thick steel inner shell. The overpack is also fitted with a very heavy lid that is secured to the overpack shell. The top lid of the overpack is approximately 9.5 inches thick, comprised of 1-inch-thick lid cover plate (top), 1-inch-thick lid shear ring (bottom), with at least 7.5 inches of concrete in between the plates. The overpack is bolted to the ISFSI concrete pad and 16 anchor studs (bolts) are used to secure each overpack. Each anchor stud is approximately 2.0 inches in diameter and 18-inches long. Each canister resides inside an overpack vessel.

ISFSI Canister Inspection Procedures reviewed by the FFT

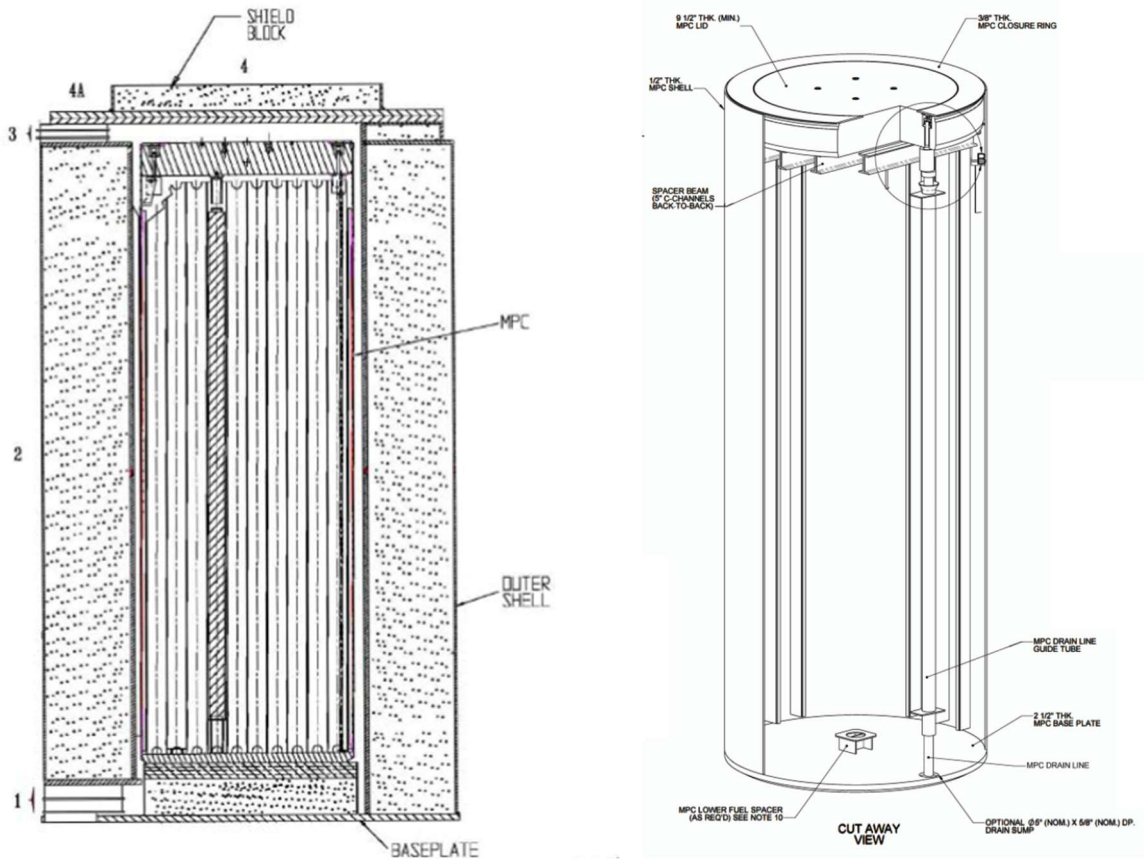
- Diablo Canyon ISFSI Multi-Purpose Canister Aging Management Program, PEP DF-23, Revision 2
- Visual Examination of ISFSI Components and the Cask Transporter, NDE VT 3-3, Revision 4
- Diablo Canyon ISFSI Overpack Aging Management Program, PEP DF-25, Revision 3

Typical Multi-Purpose Canister and Overpack

From: DCPD ISFSI FSAR Figure 7.3.1 Cross Section View of MPC, and FSAR Update Figure 4.2-13 MPC-32 Assembly Sheet 2 of 7 Revision 11, Aug. 2025

Item Description Annotations:

1. Entrance for Cooling Air space between the MPC and Concrete Overpack (4 ea.)
2. Exterior Overpack steel shield wall with 27 ½" interior concrete shield filler
3. Exit for Cooling Air heat removal (4 ea.)
4. Top of Overpack Concrete shield block and 4A steel rings
5. MPC can be used for storage and transport.



The FFT was provided with the Plant Engineering Procedure (PEP) for the Multi-Purpose Canister (MPC) Aging Management Program (AMP), dated July 2026. The document described the scope of the MPC AMP, as required by the ISFSI license renewal (which was issued by the NRC in 2025 approving DC ISFSI operation through 2064), and in the Diablo Canyon ISFSI UFSAR, Section 9.4.3.3.2. The PEP relied also on EPRI assessment criteria, ASME code cases, and industry Operational Experience (OE). The AMP monitors the accessible external surfaces of the MPC for loss of material (corrosion) and cracking of welds in the heat-affected zones from welding and the rest of the accessible canister surface. The Aging Management Review (Table 3-3 in the License Renewal Application) identified stress-corrosion cracking and pitting and crevice corrosion as degradation mechanisms for the welded stainless steel of the MPC and its closure ring, and these

degradation mechanisms are to be monitored as per the AMP. This monitoring allows for early identification of challenges to the MPC functions of confinement, structural integrity, shielding and heat transfer. Each of the canister's overpacks is also inspected for defects, including the overpack coated carbon steel, which is inspected for material loss, degradation, and degradation of the coating. The PEP DF-25 Procedure provides guidelines for the overpack inspection, including illumination requirements and validation of illumination requirements.

Visual Examinations

The visual examinations performed were ASME Code Section XI VT-3, and in specific areas identified by the VT-3 examination, additional ASME Code Section XI VT-1 examinations were performed. Upon comparison of results with acceptance criteria and comparison with results from the previous inspection for conclusions about time-evolution of observed degradation, the results of this inspection can call for supplemental examination (surface or volumetric) and/or degradation analysis, or updating the scope or frequency of the inspection plan, or other corrective actions, or none of the above. The FFT observed only the execution of the PEP; the interpretation of the inspection results will be documented in a subsequent report which was outside of the scope of the present FF.

The visual examinations were done using a robot provided and operated by the company RTT, performed by a certified inspector from the company IAG, while in parallel observed by a Level VT-3 certified PG&E employee, by Holtec who was the manufacturer of the MPC, and by NRC (per established NRC ISFSI monitoring program), and, on this occasion, also observed by the DCISC FFT. The PEP defines and assigns several responsibilities required for the execution of the PEP, and the FFT observed the execution of these responsibilities. The FFT commends PG&E for excellent planning and management of the safe and effective execution of these responsibilities. The coordination and communication, observed in pre-job briefs, over the course of a day of inspection, and in end-of-day debrief was clear, effectively engaging of all those involved in the job, including PG&E and non-PG&E individuals. A culture of supporting everyone in achieving excellence in execution of the work and providing opportunities for questions and comments was observed.

The RTT robots magnetically attach to the overpack metal, which is made of coated carbon steel, and roll up and down, and radially, giving visual access to the entirety of the MPC, and in the inner wall of the overpack. The RTT robot is equipped with VT-1 and VT-3 lighting, cameras, and stereoscopic cameras. The gap between the MPC and the overpack is designed for air flow. Air enters through three bottom (ground-level) air inlets and exits through three air outlets at the top of the overpack. The air flow ensures cooling of the canister, which continues to produce heat from radioactive decay of the elements in the spent fuel. The RTT robots are inserted through the air outlets. To achieve this, a metal grid, designed for shielding, is temporarily removed. Radiation protection teams verify the dose after removal of the metal grid and provide positioning guidelines to ensure that worker doses are not exceeded. The FFT noted that radiological monitoring at the outlet vent was also performed periodically. Any such monitoring could detect failure of the MPC confinement function of any MPC containing failed fuel element, which in turn would call for corrective actions such as returning the MPC to the Spent Fuel Pool.

The FFT noted that Level VT-3 certification of the PG&E employee and presence of such a certified personnel member during the inspection was not required, and the FFT commend PG&E for going above and beyond in providing the highest level of relevant expertise in the performance of this inspection. The FFT noted also that the industry does not currently have multiple vendors that could also provide these services. The inspection interval is sufficiently long to allow for alternate planning, where the industry options for such services to become limiting or insufficient for what is needed for Diablo Canyon ISFSI. The presence of a PG&E Level VT-3 certified employee at the inspection is evidence of this redundancy in capability, and for PG&Es readiness to provide long-term stewardship of the ISFSI. The FFT commended PG&E for this forward-looking responsible approach to long-term stewardship of the spent-fuel storage facility.

Monitoring of air intake vents

The FFT also noted that the air intake was monitored daily for obstructions. Daily Nuclear Operator rounds required a walk-down of each overpack on the ISFSI pad to identify any obstructions of the air inlet vents. If a vent has more than 50% of the air passage blocked, then the technical specification requires for the air passage to be restored within 24 hours. The FFT had the opportunity to accompany an operator and an operator in training on an ISFSI walkdown. They were both able to explain to the FFT the procedures and acceptance criteria for the walkdown. When asked why such a walkdown is necessary, the immediate response continued to be focused on execution and acceptance criteria. Upon more extended discussion, it was apparent that the operator in The FFT viewed this as an example of execution being operationalized with excellence. For example, pre-job briefs consistently review the steps that are to be executed and safety considerations. In contrast, purpose often remains implicit and is not operationalized to the same extent as safety and execution excellence. For example, pre-job briefs do not routinely restate why a job is being performed, which may limit reinforcement of the underlying purpose and importance of the procedure.

Robotic arm with Emory cloth

One of the RTT robots was equipped with an “scrubbing” pad with Emory cloth; this was to be tested as an approach to removing observed deposits, such that the metal underneath could be inspected for degradation. There was no prior industry experience with such an approach, and the team was prepared to test this new approach as appropriate opportunities arose. The Emory cloth is softer than the base metal, such that it would not induce scratches on the MPC stainless steel. This initiative is a good example of continuous improvement based on learnings from prior inspections.

PG&E staff reported that NDE experts identified rust-colored spots on the axial and circumferential welds of MPC #170 during the inspection process. Initially, it was uncertain whether the Emory cloth scrub pad mounted on the robotic inspection system would be effective in removing the deposits. To evaluate the condition, inspectors obtained an initial height measurement (h1) of the most significant deposit, which measured 0.002 inches above the surrounding surface. The robotic brush arm was then deployed and used to scrub the area, after which a second measurement (h2) showed the deposit height had been reduced to 0.001 inches. Following replacement of the scrub pad, a second brushing was performed and a third

measurement (h3) indicated a flat surface, confirming that the deposits had been completely removed. Stereoscopic camera images and depth measurements taken throughout the process demonstrated that the material was a superficial surface deposit rather than degradation of the canister itself. The cleaning allowed direct inspection of the underlying metal, which revealed no cracks or other signs of degradation. According to the NDE experts, the rust-colored residue may have originated during fabrication at Holtec's manufacturing facility if a carbon steel brush, rather than a stainless-steel brush, was used to remove weld slag and debris, leaving residual material that later rusted. PG&E retained the scrub pads for further evaluation and operational learning.

FFT Observations at ISFSI

The FFT had to obtain radiological dose monitoring devices from on-site dosimetry to measure any radiation the team was exposed to while on the ISFSI pad. The team's allowable occupational dose for the planned observation activities was 10 mrem cumulative doses, and no more than 20 mrem/hour dose rate at any point in time. The team's individual exposure during our stay with the ISFSI inspection team was under 1 millirem, and the team's dose rate remained under 2.4 mrem/hour at any given point in time while on the ISFSI pad. Radiation exposure generally scales with proximity to canisters.

After obtaining the proper dosimetry, the FFT was told there was a problem with the in-use canister inspection camera and it was in the process of being returned to service for canister inspections. About an hour later, the team was allowed to gain access to the inside of the ISFSI fence where the inspection team had set up an inspection station. Overall, the ISFSI inspection team had a few issues with the cameras and cables while the FFT was on site. The issues were: initial camera placement, electronic noise affecting screen clarity, and a need for a cable change. These issues were resolved quickly and were discussed at the daily post job briefing conducted after the work area was secured for the night.

The inspection team had set up a makeshift tent to reduce the glare of the sun, that also allowed several people to be inside the tent while the inspections continued. Inside the tent, there were two large computer screens, one of which showed clearly what the camera was recording. The front side of the camera was slowly lowered from the top air vent down between the canister and the overpack shielding. The camera had magnetic wheels that are attracted to the overpack steel inner casing so the operator can stop, speed up, or slow down the rate of camera movement. The second screen was displaying the back side of the camera to help in navigation and to be used as a second check for any potential second option on an unusual appearance of the outer canister body. It also was used to identify the exact location of the camera to avoid interferences on a support or other internal member.

The lighting and the clarity of the image was repeatedly validated using standards with a frequency of eight hours. The stereoscopic cameras were capable of identifying thickness and depth of deposits and scratches of 0.001 inches or smaller, on the surface of the 0.5-inch-thick MPC. The VT-1 and VT-3 inspections follow ASME-certified procedures, are performed by certified inspectors, and are identified in the Aging Management Report prepared by PG&E and reviewed and approved by the NRC as appropriate for early identification of degradation mechanisms that might begin to challenge the function of the MPC. Furthermore PG&E and industry OE is

proactively used to continuously improve ISFSI aging management. More than 30 ISFSI sites in the US have Holtec's MPC-32 canisters, some of them also operating in a marine environment. On this basis, FFT concluded that oversight and continuous improvement of DCP's ISFSI AMP was appropriate to ensure confidence in the safe operation of the ISFSI facility.

The inspections were slow and methodical, and there were a few signs of machining marks on the stainless-steel outer canister walls, as expected. The certified visual inspectors were both viewing the camera feed from inside the gap between the inner overpack and the outer canister surface. One inspector was stationed at the air vent with a smaller camera screen, while the other inspector was stationed inside the tent containing the two large viewing screens. As the camera progressed, either of the two inspectors could call a stop and reverse to examine a potential flaw. The two inspectors worked like a team and had great respect for each other as they viewed the outer surface of the canister being examined. The canister inspection team was very helpful to the FFT in answering our questions.

Both the DCISC Member and Consultant had ample opportunities to sit in the tent and observe the examination process. The FFT noted that the NRC inspectors were also spending time in the tent, as well as directly observing the inspection team manipulating the camera through the upper air vent. Both NRC inspectors had significant experience observing other utilities' inspection campaigns.

During the FFT observation period, the inspection team identified some dried residual rainwater deposits that were evident near the top air vent area. This water did not concern the Visual Inspectors as this has been observed at other ISFSI during their inspections. A primary area for potential flaws found infrequently is on the canister seam welds, and these were inspected carefully for any indications. According to the NRC inspectors, no canisters in the US have had any flaws or cracks reported in the inspections conducted to date.

A subsequent PG&E report with the final interpretation of the collected visual inspection images and videos was to follow. The FFT recommended that the DCISC should follow up with review of the final inspection report. During these activities, the Team was able to methodically inspect the exterior of the MPC using cameras that were designed, built, and certified to conduct VT-1 and VT-3 visual examinations. Overall, the Inspection Team was pleased with the inspection results for the eight inspected MPCs. Images were shared by PG&E with the FFT team of one of the closure ring welds. The FFT was able to observe in real time the inspection of canister welds. Thus, it was observed by the FFT that the crawler has access to inspect both the factory canister welds and the closure ring welds.

The FFT commends PG&E for the execution of the inspections. The FFT had full access to all participants in the inspections, and everyone was knowledgeable and fully transparent about their work and the limitations of their work. The FFT was able to observe that appropriate resources and expertise were available to ensure correct completion, documentation and oversight of the Inspection.

The FFT also noted the following observations regarding the DCPD ISFSI equipment and operations:

- License renewal includes four different types of canisters, including intact damaged fuel, fuel debris, and non-fuel core components; currently only intact fuel MPC-32 is at ISFSI. Future MPC-37 could be licensed, which would fit 37 assemblies, and hence 69 casks, rather than 80 casks, from expected spent fuel for operations at Unit 1 up to 2044 and Unit 2 to 2045.
- The last MPC loading was in 2024. The next MPCs will be loaded in 2027 and 2029. Loading campaigns are planned to manage heat loading in each canister, and inventory of different ages of spent fuel in the spent fuel pool. Current MPC loading campaigns are planned based on operation through 2030.
- Design life of MPC is an initial 60 years, likely extendable to 100 years or more, with AMPs to monitor for evidence of evolving degradation. Upon degradation detection, options will exist for repairs, such as cold spray and overlay welding.
- Level 2 VT recertification exams are conducted every 3 years and Level 3 VT recertification exams are conducted every 5 years using an American Society for Non-Destructive Testing (ASNT) method exam and PG&E-specific exam.
- A qualified supplier list is maintained and all suppliers must be subject to PG&E Quality Verification program audits.
- The ISFSI's renewed license requires inspection of one or two MPCs, depending on the procedure scaling factors, at a five-year interval. PG&E inspected eight, balancing the criteria of different steel grades used across the years by Holtec for the MPCs, different heat loadings (lower than 150 °C surface temperature makes the canister more susceptible to corrosion due to moisture, and older and/or lower burn-up fuel will have lower heat loading), and repetition of canisters from previous inspections for the next few cycles.
- Other examples of ISFSI pads with Holtec canisters located in marine/coastal environments include: Oyster Creek, St. Lucie, San Onofre, Pilgrim, Turkey Point, Calvert Cliffs, and others.

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)

Recommendation to PG&E: None.

3.3 Westinghouse 10 CFR Part 21 Reporting of Defects and Noncompliance

On August 19, 2026, the FFT met with Mr. Gordan Verdin, Senior Reactor Engineering Subject Matter Expert (SME). The subject of the meeting was a Westinghouse Part 21 Event Notification Report issued to the NRC on June 19, 2026, 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 system allows the use of facility-specific curves to determine when a fresh fuel assembly with IFBAs is acceptable for storage in the spent fuel pool. When the system adjusts for an IFBA loading different from what was used to develop the facility-specific curve, there is potential for the assessment to allow storage of fuel with IFBA loading that is not encompassed by the fuel storage criticality analysis of record.

“Currently, Westinghouse has not identified an existing condition where the system has allowed fuel to be stored in a manner that is not encompassed by the fuel storage criticality analysis of record for the potentially impacted plants listed below. The affected customers have been notified. Westinghouse expects to complete its 10 CFR Part 21 evaluation on or before September 30, 2026.

“The following plants are potentially affected: Cook, Diablo Canyon, Farley, Millstone, Point Beach, Salem, Turkey Point, Summer, South Texas, Vogtle, Wolf Creek” (Reference 6.9).

The FFT noted that the reporting of errors found is excellent practice, and such reporting should be commended and encouraged. If Westinghouse, PGE, and NRC follow through on this notification, this is the kind of activity that provides confidence in a well-functioning regulatory process and responsible industry vendors.

Based on the notification, this was not a criticality calculation error that currently impacted the plant and Westinghouse identified September 30, 2026, as an appropriate date for follow-up analysis. The topic was added to the current DCISC open items list and an inquiry was sent to PG&E on July 5, 2026, to be briefed on the effect that the Part 21 notification might have on the Spent Fuel Pool (SFP) as PG&E prepared for additional new fuel bundles to be placed in the pool in preparation for the October 2026 Unit 1 refueling outage. PG&E provided an initial response to Dr. Scarlet’s questions which was not sufficient to fully understand the actions being taken by

DCPP. As a result, the FFT requested an in-person meeting with the DCPP Reactor Engineering SME, this meeting was added to this agenda.

Mr. Verdin came well prepared for the discussion and provided technical presentations, fuel management documentation, SFP configuration maps, and applicable licensing and design references to support the review as discussed below. Among the documents discussed were:

- Westinghouse Letter NF-24-042, dated September 4, 2024, "Loading Pattern, Loading Pattern Risk Assessment, and Core Loading Plan for Unit 1 Cycle 26."
- Westinghouse Letter NF-PGE-26-018, dated April 29, 2026, "Loading Pattern, Loading Pattern Risk Assessment, and Core Loading Plan for Unit 1 Cycle 27."
- Reload Schedule and Energy Requirements (RSER) documents used by PG&E to establish design requirements for each reload cycle.
- Unit 1 SFP configuration maps, including the post-PGCR Shipment 5 SFP loading map.
- Applicable Diablo Canyon Technical Specification requirements governing spent fuel storage and new fuel placement.

Mr. Verdin explained that the Part 21 issue involved a calculation methodology associated with SFP criticality analyses. However, Diablo Canyon's current fuel inventory remains bounded by Technical Specification screening requirements that verify criticality safety without relying on the affected Westinghouse tool. Mr. Verdin noted that SFP criticality safety relies on a combination of fuel design controls, soluble boron concentration, fuel placement restrictions, and storage geometry (i.e. characteristics of neighboring assemblies).

DCPP receives fresh fuel assemblies in advance of an upcoming refueling and stores them in the SFP, interspersed with the spent fuel assemblies. The FFT reviewed the fundamental criticality controls associated with the Unit 1 SFP. The FFT found that SFP criticality is ensured by managing the soluble boron concentration in the SFP, enrichment of the fresh fuel, number and characteristics of rods with Integral Fuel Burnable Absorbers (IFBAs), and positioning of the assemblies in the SFP to intersperse fresh fuel among spent fuel.

During a typical refueling outage, Diablo Canyon replaces approximately 84 of the 193 fuel assemblies in the core. Because movement of fuel assemblies carries operational risk, refueling plans are developed to minimize unnecessary fuel handling. New fuel is placed in designated regions of the SFP in accordance with Technical Specification requirements and approved loading maps. Additional restrictions govern heat management and postulated abnormal conditions. For example, recently discharged high-decay-heat fuel assemblies cannot be placed adjacent to one another in specific configurations, and station commitments prohibit placement of hot fuel over rack support feet to preserve cooling air flow pathways. These controls support both criticality safety and thermal management objectives.

Diablo Canyon utilizes 17x17 Westinghouse fuel assemblies containing 264 fuel rods in each assembly. The enrichment of the new fuel entering the core will vary and is currently limited to a nominal enrichment of 4.90% by weight U-235 (with fabrication tolerance up to approximately 4.901%). The storage criteria for new fuel assemblies are based on enrichment limits, characteristics of neighboring assemblies in the SFP, count of IFBA rods in the assembly, and corresponding amount of absorber loaded on the IFBA rods. IFBAs consist of a zirconium diboride (ZrB_2) coating applied to fuel pellets that introduces burnable neutron absorption capability during early core operation. During operation, IFBAs reduce the peak power levels to maintain heat generation under the “Hot Pin” allowable core guidance, and are primarily depleted during the first several months of reactor operation. IFBAs are credited in SFP criticality evaluations, and the IFBA loading criteria is specified in the Technical Specifications for specific fuel enrichment levels.

Unit 1 Cycle 27 fuel, scheduled for loading during the October 2026 refueling outage, has a maximum nominal enrichment of 4.900% and a minimum of 80 IFBA rods per assembly, loaded at approximately 2.25 mg/inch, at least 124 inches over the length of the fuel rod. The Technical Specification requires fewer IFBA rods at this enrichment level than are currently scheduled for the Unit 1 fresh fuel being delivered this cycle. Consequently, the current Unit 1 fuel inventory exceeds the minimum absorbers necessary to satisfy SFP storage requirements. The FFT noted that these observed fuel characteristics provide an independent basis for confirming compliance with criticality safety criteria, separate from the details of the affected Westinghouse computational methodology.

Unit 2's next fuel shipment was expected after November 2026 and will incorporate a new fuel design. Current evaluations indicate that future analyses may increase the number of required IFBA rods credited for criticality evaluations; at the same time, the actual fuel being procured contains substantially more IFBA rods than historic minimum requirements. Any future changes will continue to be evaluated through the NRC-approved Westinghouse reload methodology before implementation.

Mr. Verdin clarified the role of the Westinghouse reload analysis methodology. The NRC-approved Westinghouse reload methodology is used to develop cycle-specific fuel designs, loading patterns, reload schedules, energy requirements, and SFP placement evaluations for future reloads. This methodology may be used when analyzing new fuel designs, future enrichment strategies, revised IFBA configurations, or operational changes that require detailed reactivity calculations.

PG&E also has in-house tools that assist in the planning of SFP layout. Consequently, the FFT concluded that the existing inventory remains bounded by established criticality criteria while Westinghouse completes disposition of the Part 21 issue. The FFT commends the proactive development of in-house tools to complement tools made available by Westinghouse, to better manage SFP planning in a manner that is specific to the DCPP SFP; such tools were mentioned to be useful towards upcoming SFP upgrades/maintenance activities.

Historical design information also reviewed during the meeting indicated that fuel regions dating back many operating cycles also met the screening criteria for criticality safety. The last Unit 1 fuel region with enrichment above 4.9% was identified as Region 23B during Cycle 21 and it met the applicable criticality screening requirements independent of the specific software tool identified in the Part 21 report.

Overall, the SFP storage racks are designed such that the effective neutron multiplication factor (k_{eff}) remains below 0.95 with approximately 900 ppm soluble boron concentration (including in the scenario of mistakenly placing two 4.9% U-235 fresh fuel rods adjacent to each other) and below 1.0 with unborated water conditions. Section 9.1.2.3.6 of the UFSAR indicates that DCPD Technical Specifications call for 2000 ppm soluble boron concentration in the SFP water. Mr. Verdin indicated that actual operating concentrations of the SFP are typically maintained around 2,100 ppm, providing additional margin beyond the licensing basis assumptions. The FFT also reviewed the historical development of the spent fuel storage racks. Certain storage locations were originally associated with “Boraflex” neutron absorber material. Due to industry-wide Boraflex degradation concerns identified over prior decades, current SFP analyses no longer rely on Boraflex credit. The FFT team verified that there is sufficient room in the SFP to accept the new fuel for U1 while meeting the positioning criteria that require alternation of fresh fuel assemblies with spent fuel assemblies in the SFP.

The FFT determined that DCPD has multiple measures for assuring SFP criticality safety. These measures include Technical Specification boron requirements, storage geometry, enrichment limits and corresponding IFBA requirements, administrative fuel placement controls, and reload design reviews. The FFT concluded that Unit 1’s SFP inventory meets and will continue to meet established criticality safety criteria independently of the specific calculation issue described in the Westinghouse Part 21 notification. The DCISC looks forward to reviewing the changes that are expected for resolution of the Part 21 “Fuel Pool Storage Criticality Calculation Error”.

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. The initial indications were that effects are small, and DCPD continued to monitor progress.

Recommendations to the DCISC or PG&E: None.

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

The DCISC FFT requested an update from PG&E regarding its position on an email communication sent to the DCISC on Friday, June 5, 2026, from Dr. Frank N. von Hippel. Dr. von Hippel is a Senior Research Physicist and Professor of Public and International Affairs emeritus at Princeton University. He is also involved in the Program on Science and Global Security at Princeton University.

As found in Enclosure 2 and Reference 6.10 to this document, Dr. von Hippel recommends that Diablo Canyon should “re-rack the SFPs to their original ‘open rack’ designs in which the spent fuel assemblies were separated enough so that no criticality will occur and in which air cooling would be possible if there were a loss of water. In their current dense-packed design, spent fuel assemblies are stored almost as densely as in reactor cores with each fuel assembly in a box of borated steel to prevent criticality.” Additionally, Dr. von Hippel’s recommended expedited transfer of the spent fuel to the ISFSI as quickly as possible, generally after five years of cooling time.

As a result of receiving the recommendations, the DCISC requested DCP’s position on Dr. von Hippel’s recommendations for re-racking the SFP and implementing a proposed expedited transfer of spent fuel to the ISFSI. DCP’s response can be found in this document residing in Enclosure 3, “DCP’s Response to Dr. von Hippel’s recommendations concerning re-racking the SFPs and expedited transfer of spent fuel to the ISFSI.”

In summary, DCP’s position is as follows: During DCP Decommissioning Planning in 2018-2022, PG&E conducted detailed reviews of the holistic spent fuel storage strategy, offload timeframes from the SFPs to the onsite ISFSI, and the associated risk. Consistent with NRC analysis conclusions, PG&E’s current position remains that spent fuel is safely stored in both the SFPs and at the Diablo Canyon ISFSI. PG&E currently plans to conduct spent fuel transfers to the ISFSI (via “offload campaigns”) on a nominal two-year frequency to maintain available SFP space for continued DCP operations. This frequency will be reevaluated if the state approves legislation to operate DCP past 2030.

Based on the existing SFPs design, operational procedures, previously conducted PG&E and UCLA evaluations, and NRC conclusions, PG&E has no plans to return the SFPs to “open racking” nor to transfer spent fuel to the DC ISFSI only because it has been stored for greater than five years.

Conclusions: In the past, PG&E planned to conduct an “expedited spent fuel transfer” for decommissioning, and this approach was to minimize the overall decommissioning schedule and resulting cost to ratepayers. This expedited spent fuel transfer was not to address a safety concern since PG&E, UCLA, and NRC conclusions all demonstrated that spent fuel can be safely stored in either the SFPs or the ISFSI. PG&E currently plans to conduct offload campaigns on a nominal, non-expedited two-year frequency to maintain available Spent Fuel Pool space for continued DCP operations. This frequency will be reevaluated if the State approves legislation to operate DCP beyond 2030.

Recommendations 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)

Recommendations to PG&E: None.

3.5 Meet with Nuclear Regulatory Commission (NRC) Resident Inspector

The DCISC FFT met with Mahdi Hayes, NRC Senior Resident Inspector, for an update. The DCISC last met with the NRC Resident Inspectors during its late July 2026 Fact-Finding Meeting (Reference 6.11), when it concluded the following:

The meeting with the NRC Resident Inspector was beneficial, and the DCISC should continue the meetings.

The participants discussed the following items:

- The return of the NRC Senior Resident, Mahdi Hayes, to DCPD from his recent temporary assignment at the Region IV office.
- 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. This was especially true if the session can be conducted in the late afternoon or evening session.
- 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 headquarters 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.

The FFT also engaged the Senior Resident Inspector regarding a jacket water pump leak from the telltale drain from Emergency Diesel Generator (EDG) 2-2 (See SAPN 51300035). The leak (about 0.3 gph) was active only during EDG 2-2 operation and was indicative of minor seal degradation on the pump. Initially, the leak was reviewed and determined to be acceptable for continued operability provided that a compensatory measure was established to check jacket water tank level every two hours during EDG 2-2 operation and replenish the coolant if tank level fell below 12 inches (eight gallons).

As the staff continued to pursue the EDG 2-2 jacket leak repair, further investigation into the existing guidance of the procedures associated with cooling water head tank, operator rounds, annunciator guidance, and the small leakage amount, it was determined that a new Compensatory Measure was not necessary. This conclusion was outlined in Task 10 of SAPN 51300035.

Mr. Hayes reviewed the Task 10 with the FFM team and agreed that the call for an initial Compensatory Measure was not required, given the existing procedures would cause the water tank to alarm if an annunciator signaled a low tank level. The steps in Task 10 of the SAPN provide sufficient guidance for an operator to react and take action to fill the tank with additional water to restore proper water levels in the tank.

Conclusions: The meeting with the NRC Senior Resident Inspector was beneficial, and the DCISC should continue the periodic meetings.

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-3.5)*

Recommendations to PG&E: None.

3.6 Meet with DCPD Officer

The DCISC FFT met with Blair Jones, Vice President Business and Technical Services, for a general update. The DCISC last met with a DCPD officer during its late July 2026 Fact-Finding Meeting (Reference 6.12), when it concluded the following:

The regular meetings with DCPD Officers and Directors continue to be beneficial for both organizations.

The participants discussed the current meeting agenda items and other items of mutual interest.

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

Recommendations to the DCISC or PG&E: None.

3.7 Flow Accelerated Corrosion Program

The FFT met with Chris Beard, Senior Flow Accelerated Corrosion (FAC) Engineer, for an update on the health of the station's Flow Accelerated Corrosion (FAC) Program. DCISC last reviewed the FAC Program during its May 2022 Fact-Finding Meeting (Reference 6.13) when it concluded the following:

DCPD continues to manage its Flow Accelerated Corrosion Program effectively. The numbers of inspections and replacements performed as a part of the Program were now relatively low historically and will continue be low as DCPD approaches the cessation of power operations.

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. Susceptible components include carbon steel and

low alloy steel piping, tubing, and vessels exposed to flowing water or wet steam particularly at bends, elbows, tees, and any other location where turbulent flow exists.

The objective of the FAC Program is to provide a high degree of confidence against rupture of FAC-susceptible components, and most susceptible piping is located in non-safety-related systems. The program includes identification of susceptible systems, predictive modeling, incorporation of plant and industry operating experience, ultrasonic thickness measurements, acceptance criteria, repair and replacement strategies, and qualification requirements for FAC program engineers. Inspection data are entered into the "CHECWORKS" software system, an Electric Power Research Institute (EPRI) -developed predictive modeling program that tracks degradation rates, projects future wall thinning, and identifies locations requiring additional inspections or replacement. DCPD personnel indicated that CHECWORKS is currently undergoing transition to a web-based version that is in beta testing. DCPD personnel also remain active participants in the CHECWORKS Users Group, an industry forum sponsored by EPRI that meets approximately every six months to exchange operating experience and program improvements.

Mr. Beard reported that DCPD continues to employ a proactive replacement philosophy designed to eliminate susceptible materials before significant degradation occurs. Rather than replacing isolated degraded sections, engineering evaluations often support replacement from "source to sink" using more FAC-resistant materials throughout an entire piping run. This approach allows adequate time for engineering, design, procurement, and installation planning and helps avoid emergent repairs, which can place significant demands on plant organizations. DCPD management has remained supportive of these long-term replacement strategies, particularly where material upgrades can permanently reduce FAC susceptibility. Examples discussed included previous replacement of high-pressure extraction steam piping, feedwater piping, condensate polisher piping, and replacement of the steam generator blowdown tank.

Discussion focused on several current FAC monitoring activities. Vacuum leaks and air in-leakage into the condenser are monitored through helium leak detection techniques and observation of non-condensable gas removal systems. Particular attention has been directed toward small-bore piping, generally defined as piping less than two inches in diameter, because higher fluid velocities increase susceptibility to FAC. Mr. Beard noted that small-bore piping accounts for many FAC-related replacements due to its combination of high flow rates and relatively thin wall sections.

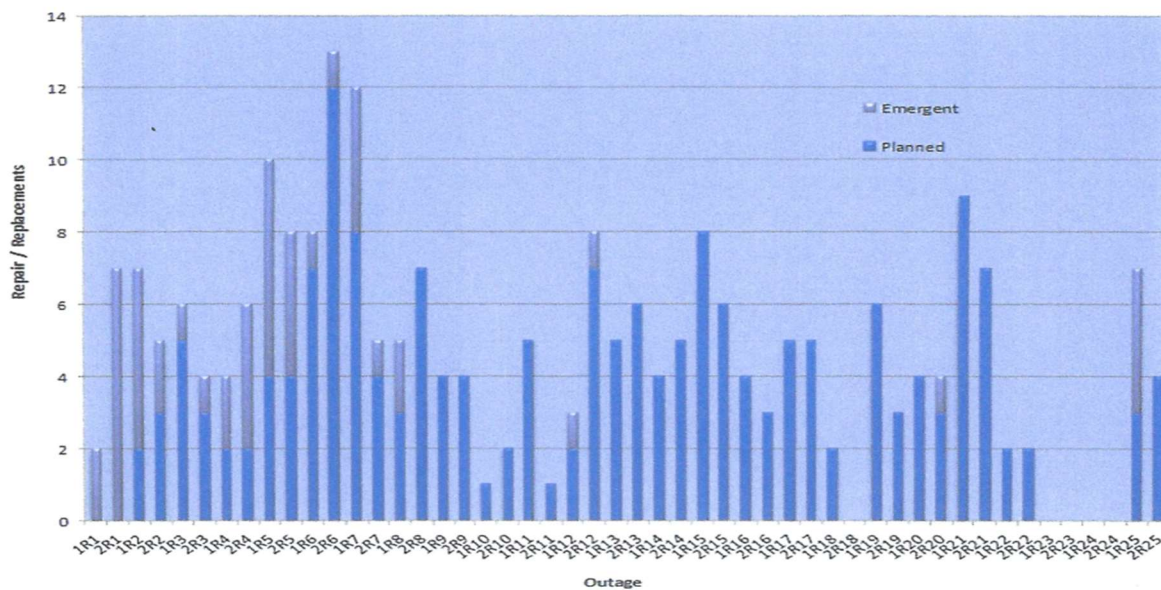
The FFT also discussed inspection and replacement activities involving feedwater heaters, including both pressure-boundary and non-pressure-boundary components such as tube supports. Recent inspections of low-pressure feedwater heaters located near the condenser utilized ultrasonic testing and chromium verification testing. These inspections confirmed that the components remain in very good condition from a FAC perspective. Stainless steel lagging installed on exterior surfaces has also helped prevent direct steam impingement and associated damage mechanisms. During Refueling Outage 2R25, lagging was removed from selected locations to verify chromium content, confirming that chromium-bearing materials had been installed as expected.

A significant factor contributing to DCPD's relatively favorable FAC experience is the use of chromium-containing steels. Previous DCISC reviews documented that Unit 2 piping was

fabricated using material containing approximately 0.4 percent chromium, whereas Unit 1 piping contained little or no chromium despite similar material specifications. Although not recognized at the time of plant construction, this chromium content substantially improves resistance to FAC. The plant continues to perform material verification testing to identify locations containing chromium and to target replacements where lower-alloy materials remain in service. Engineers also discussed the "entrance effect," wherein FAC rates can increase at transitions from chromium-containing materials to carbon steel with little or no chromium.

The graphic below provides an outage history for DCPD Units 1 and 2 FAC Program since the beginning of plant operations. The initial plan of shutting down units 1 & 2 at the end of 1R24 and 2R24 is indicated by a lack of emergent/planned repair or replacement for outages 1R23 through 2R24. The subsequent decision to support an extended license for both units by the company, the state regulators, and subsequent NRC approval of the extended license amendment for 20 additional years of operations with implementation of Aging Management Plans (AMPs) have, as a consequence, required an increase in emergent/planned repairs or replacement of piping for outages 1R25 and 2R25. This included an unexpected increase in emergent piping replacements for Unit 1. Even so, the repair and replacement rates remained no higher than historical trends. The following graph shows trends in FAC-driven repairs and replacements for both DCPD units:

FAC Repair/Replacement History for DCPD Units 1 and 2



Additional inspection and aging management activities associated with license renewal were reviewed. These included nondestructive examinations, eddy-current testing, and chemistry evaluations for replacement feedwater heater components. Newer feedwater heaters utilize Type 439 ferritic stainless-steel materials. While these materials provide improved corrosion resistance, they introduce different known degradation mechanisms, including mechanical fatigue. Personnel also described ongoing efforts to remove copper-containing materials from balance of plant systems to reduce the potential for copper transport and deposition in steam generators.

The FFT was informed that an INPO review visit focused on FAC performance is scheduled for December. Overall, DCP's FAC Program continues to emphasize predictive modeling, early detection, material replacement, and long-term planning to maintain system reliability and minimize emergent repairs. Recent completion of Refueling Outages 1R25 and 2R25 also completed the current cycle of ASME Section XI In-service Inspections, and no code-required inspections were deferred. The FFT concluded that DCP continues to maintain a mature and proactive FAC Program that remains supported by plant management, engineering organizations, and industry operating experience.

Conclusions: DCP continues to maintain a mature and proactive Flow Accelerated Corrosion Program that remains supported by plant management, engineering organizations, and industry operating experience.

Recommendations to the DCISC or PG&E: None.

3.8 Station Update

The DCISC FFT met remotely with Justin Rogers, Station Senior Director, for a Station Update. The DCISC last received a Station Update during its late July 2026 Fact-Finding Meeting (Reference 6.14), when it concluded the following:

DCP'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.

Mr. Rogers reported that both units have operated at full power since the previous DCISC update, with no major equipment issues affecting generation. The FFT discussed the following topics with Mr. Rogers:

- Power Production – Both units were operating at full power. Unit 2 was scheduled to reduce power to approximately 50 percent in the coming weeks to support circulating-water tunnel debris removal and cleaning activities. The unit will return to full power after completion of the tunnel work. DCP continues to evaluate coatings intended to reduce biofouling and organic growth on tunnel concrete surfaces under varying seasonal conditions.
- Station Performance Indicators – All performance indicators remain green except for Industrial Safety. A recent worker injury involving a broken arm caused Industrial Safety to move out of green status.
- Human Performance and Safety – No significant human performance events were reported. The industrial safety event involved an individual sustaining a broken arm after contact with a door, berm, and I-beam structure.

- Equipment Issues – No equipment issues posed a current threat to generation, although several emergent issues remained under management review. The Cause Evaluation for the Startup Transformer 2-1 bus duct failure that occurred in May 2026 was expected to receive final review and approval in late September.
- Preventive Maintenance – As a preventive measure, the station plans to replace equipment associated with Startup Transformer 1-1 to reduce the likelihood of a similar arc-strike event.
- Maintenance Outage Windows (MOWs) – Maintenance Outage Windows for the Emergency Diesel Generators were completed without significant challenges.
- Transformer Extent-of-Condition Reviews – The station completed extent-of-condition reviews for startup and unit auxiliary transformers and replaced all stress cones to support new cable terminations on Unit 2.
- Refueling Outage 1R26 – Preparations are underway for Refueling Outage 1R26, scheduled to begin October 14, 2026. The FFT observed the arrival of new fuel assemblies in the DCPD parking lot, on the morning of the emergency preparedness drill.
- Major Projects – Significant projects currently underway included: Main Control Room Annunciator System upgrade; Simulator upgrade (completed and available for operator training); Fuel Transfer System replacement; Circulating Water Pump refurbishment; Main Generator relay replacements; and Triconex system hardware upgrades.
- Fuel Transfer System Replacement – Replacement of the Fuel Transfer System began during Refueling Outage 1R25. The activity requires approximately 12 days to complete and could not be accommodated during the previous outage without affecting the critical path. Completion of this work, together with associated demolition activities, was expected to be a primary driver during the first five days of Refueling Outage 1R26.
- NRC Oversight – The most recent NRC inspection focused on the Independent Spent Fuel Storage Installation (ISFSI). The FFT interfaced with the NRC team during ISFSI inspections.

The FFT found that the update on recent station activities was informative, and the Station Senior Director was knowledgeable and transparent.

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

Recommendations to the DCISC or PG&E: None.

3.9 Evaluated Emergency Preparedness Exercise

The DCISC FFT met with Tom Jones, Senior Director of Regulatory, Environmental & Repurposing; Andy Warwick, Manager, Emergency Preparedness, and Jordan Tyman (remote), Director of Risk, for a pre-job brief on the NRC Evaluated Emergency Preparedness Exercise. The DCISC last observed an NRC Evaluated Exercise on July 31, 2024, (Reference 6.15) when it concluded the following:

The July 31, 2024, DCPD NRC-Evaluated Emergency Exercise appeared well designed and implemented. The scenario involved a sufficiently large release to require declaration of a General Emergency and to exercise Ingestion Pathways response. All exercise objectives were met.

The purpose of the 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. Because this exercise included the Federal Emergency Management Agency (FEMA) evaluation for ingestion pathways, the scenario involved a simulated release large enough to require declaration of a General Emergency. The Emergency Preparedness Exercise was designed to evaluate the proficiency of DCPD personnel in implementing the principal Emergency Plan functions in response to a radiological emergency sufficiently severe to exercise ingestion pathways responses. The Technical Support Center (TSC), Operations Support Center (OSC), Emergency Operations Facility (EOF), Unified Dose Assessment Center (UDAC), and Joint Information Center (JIC) participated. The plant simulator acted as the Control Room and drove indications in DCPD's Emergency Response Facilities.

The NRC, Sheriff's Watch Commander, County Emergency Operations Center, and the State of California Office of Emergency Services Warning Center participated along with offsite agencies within the DCPD Emergency Planning Zone as described in the County's extent of play agreement with FEMA.

The exercise had multiple objectives including 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

The Emergency Drill commenced activities shortly before 8:00 am. The FFM Team was given Blue Vests to wear for the Drill, to identify them as observers. Evaluators, and participants from different types of organizations were also wearing vests of different colors to identify their role. Preparatory meetings of the FFT with PG&E had provided the FFT with the expected scenario. On the day of the Drill, the FFT was given printed drill scenarios, which indicated the expected

response on the basis of which the drill participants were being evaluated. The FFT was instructed to safeguard the drill scenario, and to give no indication to the drill participants (through their facial expressions or movement about the space) of the timing of upcoming events. This was crucial to avoid invalidation of the drill. At the end of the drill, during the evaluated self-critique, a discussion of “unearned information” was held by each of the teams participating in the drill.

The exercise scenario is summarized below and provided activities to generate responses from the Drill participants as follows:

7:00 am Drill Setup

7:50 am Initial Conditions:

- Unit 1 and 2 in mode 1 at 100% power.
- Today is a plant holiday and staffing is reduced to essential personnel only.
- No major plant evolutions are in progress.
- Equipment is per the Ops Shift Logs.

Meteorological Conditions and Forecast

- Morning through early afternoon winds will be predominantly from the Southeast towards the Northwest. Winds will then transition from the Northeast towards the Southwest in the evening.
- Wind speeds will be predominately between 5 to 10 mph with gusts up to 25 mph
- No precipitation is expected

8:05 am **Event 1:** Auxiliary Salt Water (ASW) pump 1-1 trips on overcurrent. ASW pump 1-2 fails to start automatically. The operating crew takes action to manually restore ASW to Unit 1 per OP AP-10.

There were no Emergency Plan actions associated with this event. This event was strictly an operational response to a plant issue. No offsite assistance should be requested for this event.

8:10 am **Emergency Classification: Site Area Emergency (HS1.1)**

Event 2: Adversaries initiate an attack inside the Protected Area causing plant damage.

- Security engages the adversaries and notifies the NRC.
- The Shift Manager declares a Site Area Emergency.
- The DCPPERO are called out and directed to offsite facilities.

- 8:20 am Unit 1 Viewpoint will alarm for the letdown monitor being greater than 3 Rem/hr, which is the initial indication of fuel damage. However, the Radiation Protection Access Senior will be in duck and cover due to the hostile action and will not receive the alarm. When the offsite Emergency Response Facilities (ERFs) are activated, viewpoint data will be available to the exercise players via the Train B In-Core Temperature Monitoring System Failure.
- 08:25 am **Event 3:** A steam line fault causes a Safety Injection (SI) actuation on Unit 1.
- Residual Heat Removal (RHR) pump 1-1 fails to start on the SI signal due to a sheared shaft. RHR pump 1-1 breaker trips on overcurrent after the shaft shears due to mechanical imbalance causing contact between the rotor and the case.
- The site is in duck and cover with no movement allowed unless coordinated with DCPD security.
 - DCPD security coordinates with local law enforcement for offsite assistance and setup of an Incident Command Post.
- 08:45 am **Event 4:** Damage to the Unit 1 Refueling Water Storage Tank (RWST) starts a 10,000-gpm leak. Water leaking out of the RWST will route through the 115' RCA bench storm drains to the Diablo Creek north of the Protected Area.
- Law enforcement establishes the ICP and coordinates with DCPD security to provide offsite assistance.
- 08:50 am **Event 5:** An event results in a loss of all offsite power to both Unit 1 and Unit 2.
- DCPD ERO members begin to arrive and activate offsite response facilities. The DCISC FFT traveled to the EOF
- DCPD tech specialists are notified of events in progress and directed to report to the ICP.
 - Law enforcement arrives onsite and DCPD security provides escorts, building schematics, adversary known locations, routes of travel, and other site details.
- 09:15 am **Event 6:** Emergency Diesel Generators 1-1 and 1-3 trip due to damage caused by a saboteur.
- With all onsite adversaries neutralized, contained in the cold machine shop, or in custody, DCPD security works with the ICP to establish safe pathways

for firefighters and repair teams to respond to the plant Protected Area with DCPD security escorts

10:35 am **Emergency Classification: General Emergency (RG1.2)**

Event 7: A Steam Generator tube rupture occurs (400 gpm) on Steam Generator 1-1. This represents a loss of the Reactor Coolant System (RCS) barrier A.1. The previous steam line break on lines 1 and 2 coincident with the tube rupture results in a loss of the Containment barrier A.1. This event results in a radiological release to the environment, and the only way to stop the release is to cool down and depressurize the RCS.

- The Emergency Director at the EOF declares a General Emergency based on dose assessment results indicating doses at or beyond the site boundary exceed Protective Action Guidelines (PAGs).
- The Emergency Director declares Protective Action Recommendations (PARs) based on the General Emergency to:
 - Evacuate Protective Action Zones (PAZs) 1 & 2
 - Evacuate Ocean to five nautical miles
 - Administer Potassium Iodine (KI) per county plan
 - All other PAZs monitor and prepare
- The ERO performs dose assessment for the ongoing radiological release and determines that expanding PARs is not required.
- Offsite field monitoring teams are dispatched and directed to find the plume and conduct sampling activities.
- Potassium Iodine (KI) is issued and ingestion directed for field teams and emergency workers downwind of the release

1:00 pm Drill terminated
Individual facility critiques began
Overall critique began

The FFT started its observation activities in the site Simulator to witness the initial stages of the Emergency Drill. At the point when a Site Area Emergency was declared from the simulator, the FFT then traveled to the EOF, located on Kansas Avenue in San Luis Obispo. When the FFT arrived, 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 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.

The FFT's 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 (PARs) needed to protect the health and safety of individuals located in affected Protective Action Zones (PAZs). There are 5 PAZ's within 10 miles of the DCPD Station. The State of California has added additional PAZ's 6 through 12 that are beyond the 10-Mile federal requirement. During the course of the drill event, the emergency escalated to a point where

simulated radiation released from DCP, and a PAR was declared which required a certain PAZ to initiate specific protective measures.

Once the EOF positions were staffed, the EOF Emergency Director (ED) became the leader of the Emergency Response Team who had the responsibility to issue Emergency Classification upgrades and recommend PARs and the associated PAZs as needed. In order to assess the need for an Emergency Classification upgrade, the EOF ED confers with the TSC for plant conditions and also with the UDAC for radiological conditions that may require additional PARs in an affected PAZ. The communication amongst the EP Team are vital for correct assessment of plant and radiological conditions.

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 local media and National TV correspondents asking questions. There were also numerous written news releases in English and Spanish, by PG&E, SLO County and other local officials. (As this was a drill, no news releases were actually released.)

The FFT that participated in the evaluated exercise was supplemented by a Risk Communications expert, and DCISC Special consultant, Dr. Timothy Sellnow, of Clemson University, who observed the EOF/UDAC/SLO County facilities as well as activities in the JIC location. The JIC building contains a large press room and additional conference rooms that support communications to the public.

Risk Communication Observations and Analysis

Dr. Sellnow observed the simulation at the Diablo Canyon Power Plant on September 1, 2026 as well as some post-event PG&E hot-wash (briefing among PG&E employees) on September 2. The observation occurred at multiple sites throughout the event, with the following scope:

- Describing observations of the communication process from the perspective of risk communication.
- Interpreting the observations from a neutral stance to determine the alignment of communication observed and the expectations of effective risk communication.
- Evaluating the communication content shared and the interactions observed from the perspective of best practices for effective risk communication.
- Recommending risk communication strategies for addressing identified weaknesses and threats, and for capitalizing on risk communication strengths and opportunities.

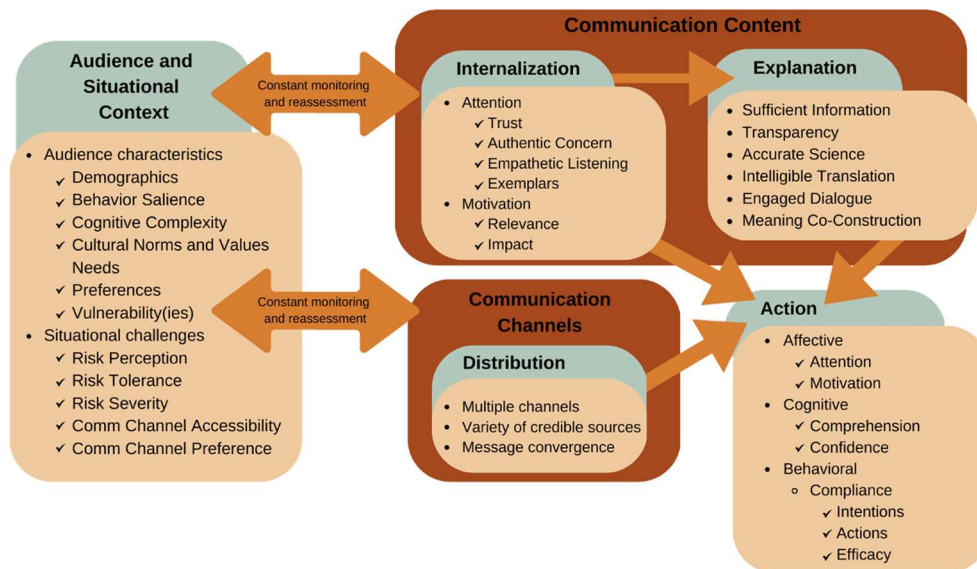
Dr. Sellnow reported that the risk communication process was well-executed in all phases of the actual simulation. Specifically, the simulation participants displayed:

- A clear knowledge of network through which information was intended to flow;
- Coordination among decision-makers within and surrounding the plant;
- Coordination among county officials; and
- A consistent and central focus on the safety of citizens in areas considered at risk.

Generally, the participants were able to maintain clarity of purpose and continuity in their execution of duties even as the simulation heightened both functional difficulty at the plant and uncertainty for the community’s safety. As is clarified in the key observations below, there were some opportunities for further development of the public communication.

The key observations listed in this section focus exclusively on instructional risk communication. This focus acknowledges that preparedness and following recommended protective actions during a radiological event are primarily instructional in nature. In this section, the theoretical context for assessing public communication is provided. This conceptual explanation is followed by several key observations explained through the structure of the “IDEA” model.

The context for assessment is based on the best practices of public-facing risk communication as summarized in the IDEA model (Sellnow et al., 2017; Reference 16) (Sellnow & Sellnow, 2024; Reference 17). The model has been applied in many risk contexts on a global level, including radiation preparedness (Kim & Lee, 2022; Reference 18). In addition to considerations of the intended audience’s situational context, the model focuses on internalization, explanation, distribution of messages, and recommendations for protective action (see Figure below).



IDEA Model Theoretical Framework for Instructional Risk and Crisis Communication

As the IDEA model summarizes, a radiological emergency requires communication to multiple audiences. Thus, the audiences must be identified and the characteristics of each must be considered in the development of emergency messages. Additionally, situational variables of the evolving emergency must also be considered. The content of the messages shared with audiences should be considered according to four broad categories: internalization, distribution, explanation, and action. Internalization focuses on alerting the audience to the specific threat they face. During a radiological event, the risk may vary for segments of the audience, depending on their location. Distribution focuses on message access and trust in the channels and sources providing the

instructional risk communication. Access may vary among segments of the audience, and trust in the source or channel may be uneven. For example, Kim and Lee (2022) found segments of their audience did not trust mass media to accurately share emergency information about a radiological emergency. Explanation is content shared that explains what is happening and why. Finally, action content provides recommendations for protecting oneself and loved ones during an emergency. These actions must be seen as efficacious or within the capacity of the audience to complete them with reasonable effort.

Key Observation for Internalization - The audience in the community surrounding the Diablo Canyon Power Plant (DCPP) is expected to have an awareness of the potential risk of a radiological event. Distribution of materials such as the calendar distributed broadly by DCPP is replete with instructional risk messages designed to promote general awareness. The challenge for DCPP is to provide content during an emergency detailing which regions surrounding the plant are facing heightened risk. Hence, the following observation is provided:

- Messages provided during the simulation showed a clear understanding of and capacity for identifying and addressing audience segments facing heightened risk due to a radiological emergency.

During the exercise, those who were at immediate risk of contamination due to the radiological plume were clearly identified and related messages were shared with them. The UDAC room has a map of the PAZs with the latest dose information, which during the drill informed timely decisions of which protective actions should be deployed and in which PAZs. Emergency preparedness education through distribution of the calendar by the County, and use of social media by PG&E has sought to create a general community awareness of their location in relation to the plant. The calendar, for example, specifically instructs each calendar recipient to identify and write in their PAZ number on page 12 of the calendar, right below a map of the PAZ and EPZs.

Key Observations for Distribution - An effective message distribution process relies on consistency across multiple channels. This message sharing process works best when it is intentionally redundant, meaning audiences find the same content distributed from multiple sources. Three observations related to distribution emerged during the DCPP simulation:

- Coordination among agencies was excellent. The press conferences featured multiple agencies sharing consistent messages. This collaboration was evident in the Joint Information Center (JIC) discussions and in the actual press conference. Consistency and accuracy were prioritized as the press conference was delayed until confirmation of all content was at a reasonable level of confidence.
- The response strategy included social media, both in the identification of rumors and distribution of information. In preparation for the press conference with the public, a PG&E employee summarized all observed incorrect information on social media, and a plan was made to ensure that those topics are covered during the press conference, and that social media messages would address the incorrect information that appeared to circulate. Social media continues to rise as a primary source of helpful information

and distracting misinformation during emergencies (Muhammed & Mathew, 2022; Reference 19). Thus, the focus on social media in the JIC is prudent.

- Maintaining redundancy in both emerging and traditional channels is recommended. Redundance remains important in emergency alerts. Although abandoning public siren systems or print items such as calendars may be tempting as cost-cutting measures, maintaining these options assures the broadest outreach. Some communities choose to rely solely on emergency alert systems through mobile phones. The problem with doing so is two-fold: 1) not everyone has equal access to cell phone technology; and 2) alert fatigue leads to a temptation for an increasing number of citizens to opt out of or ignore phone alerts (Sutton and Wood, 2025; Reference 20).

Key Observations for Explanation - Explanatory messages are evaluated based on the degree to which the technical information is translated to a level non-scientists can easily comprehend and their capacity to satisfy the curiosity of the audience. Two observations for explanation emerged during the exercise:

- The information provided to the publics relied on an understanding that the audience was well-informed or savvy about the potential for a radiological emergency. The assumption that the audience is well-informed about the potential for radiological emergency is fitting, however, it is based on consistent public education and opportunities for public interaction with county and industry representatives. Therefore, continued emphasis on public education and interaction as part of emergency preparedness is essential to an effective response by DCP.
- Recommendations regarding agriculture require more explanation. Recommendations about animals and open water sources exposed to the radiological plume were provided with limited explanation. This lack of explanation could result in concerns expressed by the greater audience about safety in the food supply. More explanation about how concerns over agriculture are limited to precise areas may be warranted.

Key Observations for Action - Motivating audiences to engage in protective action is dependent on their clear understanding of what they need to do and their belief/confidence that they have the ability and resources (efficacy) to achieve the recommended actions. Two observations for action emerged during the emergency simulation:

- As stated during the exercise, the call for evacuation would likely foster alarm among parents and perceived helplessness among those with limited resources. Residents of the community likely have an understanding that radiological emergencies can occur. A concern is that “moving” students from a school in an at-risk zone to a safer location may create alarm for parents. Additional information about how parents can or are not allowed to respond during such movement is warranted. Moreover, those who lack transportation or who have special needs may require additional clarification or advice on how to reach out to emergency responders to assist in their evacuation.

- Facilities housing incarcerated people would benefit from additional interaction with emergency responders. Given that facilities with incarcerated inmates are limited to sheltering in place, additional information and consideration of capacity for protecting inmates and staff during a radiological event is warranted.

In general, the communication response provided by community agencies and DCPD was effective during the exercise. The drill included simulated members of the public (simulated by PG&E drill participants), thus by the nature of the drill, it did not afford the opportunity to assess the actual effectiveness of risk communication in the relevant communities. For example, effectiveness of actually evacuating all people within the expected time window, through a concerted effort of public communication and first responder actions was part of the scope. The drill only provided the FFT the opportunity to observe inter-agency interactions, and the effectiveness of planning for public communication. As the observations listed above explain, there are opportunities for continued success and improvement. Continued public education and access to multiple forms of information, including traditional forms of communication, is needed. An effective response depends on having an informed community before an event occurs. There is a need for additional information about the threat a radiological plume creates for agriculture. There is also a need for further information directed to parents of children who may be moved during an emergency and for those who may find evacuation difficult. Finally, further communication with facilities housing incarcerated people is needed.

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 upon. The UDAC Team operated in a unified manner across agencies and areas of expertise, the radiological data was being analyzed and recommendations were discussed with each of the other teams. The SLO County team was effective in their deliberations and recommendations, communicating frequently with the other Teams at the EOF. SLO County staff indicated they also had access to other heavy equipment if needed.

The PG&E critique went over exercise objectives, lessons learned, and actional items. Across all members of the PG&E team participating in the critique there was an evident culture of seeking to learn and continuously improve processes, available equipment, and training. During the evaluated self-evaluation among all drill participants, while each sub-team that the FFT observed during the self-critique had a distinctly different culture of communication and engagement with one another, similarly candid feedback was provided and assigned leaders and note takers ensured that all action items were captured so that they can be followed up upon. The FFT observed one instance of colleagues going above and beyond in helping each other with note-taking responsibilities to ensure that no time was lost in recording all actionable items identified during the critique.

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 Preparedness Exercise critique and NRC and FEMA evaluations in a future fact-finding meeting when they are completed. (D9/26FF-3.9)

Recommendations to PG&E: None

3.10 Emergency Preparedness Meetings

The FFT met with representatives from San Luis Obispo County Office of Emergency Services (SLO County OES), California Office of Emergency Services (Cal OES), and the California Men's Colony (CMC) to discuss emergency preparedness. The FFT reviewed the current DCPD Emergency Plan and the Emergency Preparedness Calendar. These documents describe twelve Protective Action Zones (PAZs) and three Public Education Zones (PEZs). Below is a summary of the documents reviewed:

- PG&E NRC-FEMA Evaluated Drill (September 2, 2026) documents: the exercise scenario package, controller/evaluator materials, and exercise objectives.
- PG&E Emergency Plan Update to NRC Revision 5.00 (December 13, 2025)
- NRC-FEMA Memorandum of Understanding (MOU) (July 1, 2024)
- FEMA to Cal OES Reaffirmation of Limiting Federal Evaluation to PAZs 1-5 (July 5, 2023)
- California-FEMA EPZ Reaffirmation Agreement (June 26, 2023)
- FEMA Region IX to NRC Region IV Adoption of Proposal for EPZ Review (May 16, 2000)
- FEMA Proposal to Limit Plan Review and Exercise Evaluation Requirements at DCPD, assigning primary responsibility for PAZs 6-12 to the State of California (August 12, 1998)
- FEMA Program Manual

During Fact-Finding Meeting activities conducted in August/September 2025, December 2025, and March 2026, representatives from PG&E and Kelly van Buren of SLO County OES explained the distinction between PAZs 1-5 and PAZs 6-12. PAZs 1-5 are located within the NRC-mandated 10-mile plume exposure emergency planning zone and are subject to NRC and FEMA oversight, including evaluated emergency preparedness exercises. PAZs 6-12 extend beyond the NRC-required 10-mile radius. The FFT reviewed several documents describing the regulatory framework governing emergency preparedness outside the NRC-required emergency planning zone. These documents indicate that FEMA and the NRC limit formal plan review and evaluated exercise requirements to PAZs 1-5. Consistent with this framework, FEMA's 2023 reaffirmation to Cal OES states that "Public Education Zones 13 through 15 are outside of the plume exposure pathway and are not included in State or Federal plan review." Similarly, the California-FEMA reaffirmation agreement states that California will assist the County in coordinating and reviewing PAZs 6 through 12 and that FEMA may provide courtesy reviews upon County request.

The NRC-FEMA MOU (July 2024) indicated that licensees maintain agreements with offsite organizations to provide emergency onsite support services. Consistent with these regulatory requirements, the DCPD maintains Letters of Agreement that are reviewed annually with the following supporting organizations: Cambria Community Healthcare District, CAL FIRE / San Luis Obispo County Fire Department, California Highway Patrol, DOE Oak Ridge Operations Office, French Hospital Medical Center, Institute of Nuclear Power Operations (INPO) Nuclear Power Plant Agreement, Marian Regional Medical Center, PG&E Company Helicopter Operations Department, San Luis Ambulance Service, and United States Coast Guard.

Based on the information reviewed by the FFT, Emergency Preparedness Exercises such as the NRC/FEMA-evaluated exercises are routinely conducted within PAZs 1-5. The FFT did not identify comparable evaluated exercises for PAZs 6-12, except for specific function pertaining to infrastructure common to all PAZ 1-12; for example, the French Hospital Medical Center is in PAZ 8, and it is designated as an offsite medical response organization in the Emergency Preparedness Program; in this role, it is evaluated by FEMA as part of the capability to receive, treat, and manage contaminated or injured individuals following a radiological incident. Otherwise, responsibility for preparedness planning in PAZ 6-12 resides primarily with SLO County emergency management organizations, with support available from Cal OES and FEMA upon request.

The Fact-Finding Team also reviewed guidance regarding public notification and communication to transient populations. Guidance in the FEMA Program Manual calls for:

“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.” [...] “Provisions for providing accessible public information for those with access and functional needs” [...] “Accessibility and Foreign Language Translation of Public Information Materials”

The DCPD Emergency Plan states that siren information stickers are distributed to businesses, hotels, motels, places of public assembly, parks, and scenic locations within the emergency planning zone, and that businesses serving transient populations are provided information describing appropriate actions to take if emergency sirens sound. PG&E provided a photo of the siren sticker (shown below), and a list of San Luis Obispo County radio and television stations supporting DCPD Emergency Management, in English and Spanish (shown below).

Local Radio & Television Stations

<u>ENGLISH TELEVISION STATIONS</u>		<u>SPANISH TELEVISION STATIONS</u>	
KEYT	Channel 3	KTAS	Channel 33
KSBY	Channel 4/6	KPMR	Channel 38
KKFK	Channel 11		
KCOY	Channel 12		

<u>ENGLISH RADIO STATIONS</u>		<u>SPANISH RADIO STATIONS</u>	
KVEC	Channel 920 AM	KJDJ	Channel 1030 AM
KPRL	Channel 1230 AM	KLMM	Channel 94.1 FM
KKJL	Channel 1400 AM	KRTO	Channel 97.1 FM
KUHL	Channel 1440 AM	KLUN	Channel 103.1 FM
KCBX	Channel 90.1 FM	KIDI	Channel 105.1 FM
KZOZ	Channel 93.3 FM	KSMY	Channel 106.7 FM
KPYG	Channel 94.9 FM		
KXTZ	Channel 95.3 FM		
KSLY	Channel 96.1 FM		
KSTT	Channel 104.5 FM		
KJUG	Channel 98.1 FM		

DCPD Siren Sticker and Radio and Television Station List

PG&E also indicated that social media platforms are used for disseminating information during a radiological event and this functionality was employed during the September Evaluated Exercise. During fact-finding activities, Committee members did not observe examples of this information at restaurants, cafés, hotels, or other transient-oriented facilities that they visited. In several instances, hotel personnel appeared to have limited or inconsistent knowledge regarding radiological emergency preparedness. These observations are presented only as examples and are not intended to represent a comprehensive assessment of public education effectiveness. Nevertheless, the FFT concluded that the effectiveness of public communication and education directed toward transient populations, particularly as described in Section 8.2.3 of the DCPD Emergency Plan, warrants additional review by DCISC.

Conclusions: 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.

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 DCPD

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)

Recommendations to PG&E: None.

3.11 Offsite Emergency Protective Action Measures

The FFT visited California Men's Colony (CMC) and met with Captain Mario Fernandez on September 3, 2026. In the past, DCISC representatives have had discussions with SLO County OES and CMC exploring the topic of sheltering in place as a protective measure in case of an emergency. The meeting was also attended by Harold Powell, Fire Chief, and Christine Freiker, Associate Warden of Control Operations. The purpose of the meeting was to better understand the interface between DCPD Emergency Preparedness and the Emergency Plans for CMC, which is located in Protective Action Zone 8 (PAZ 8).

The following topics were discussed:

- The radiological incident procedure within the CMC Emergency Operations Plan
- How sheltering in place measures would be implemented for the spaces where workers and inmates would be during a sheltering in place event, and whether the available infrastructure appropriate for sealing windows and doors, turning off ventilation, having access to communications, access to potassium iodide (KI) are in place
- Whether the staff is trained on the protocols for such a scenario, and whether there are periodic drills or exercises
- What support, resources, and training are available to CMC to implement radiological incident preparedness
- What communication channels are available to CMC (staff and inmates) during a radiological incident scenario, and whether these communication channels are suitable

Emergency Preparedness

CMC has an Institutional Emergency Operation Plan for All Hazards, which includes a plan for Radiological Incidents, following the California Code of Regulations (CCR) and Board of State and Community Correction (BSCC) Title 15 – Operational Regulations, and Title 24 – Building Standards. The Radiological Incidents Operational Plan is renewed every year, and acknowledged by CMC leadership (e.g. medical, operational etc.). CMC leadership is responsible for disseminating the information and relevant training on matters of Emergency Preparedness for

Radiological Incidents. Specific training for radiological incidents is not included in routine training nor in continuing professional training. CMC was invited but was unable to participate in the Emergency Drill in September 2026. CMC appreciated the invitation and wished to be invited in future emergency drills and trainings made available by the county on the topic of responding to Radiological Incidents.

In the event of a Radiological Incident, CMC coordinates with CalFire EOC in Templeton and Cal OES and SLO County OES, California Highway Patrol, SLO County Sheriff, and the SLO Police Department. For radiological decontamination, no specific provisions are in place, but overlap might exist with Hazardous Materials (HazMat) Emergency Decontamination, for which there is a mutual agreement with CalFire, for coordination, support and training.

Protective actions during a Radiological Incident can include sheltering in place, and/or administration of potassium iodide (KI). The FFT met with the on-site pharmacist who confirmed that they have on inventory a certain number of doses of KI for the entire CMC population. The specific dosing instructions were not known by memory to the pharmacist, but she indicated that she has access to instructions for KI dosing. A plan exists for sheltering in place, in a specific facility that includes an enhanced outpatient program for highly volatile inmates, and a 50-bed mental health crisis center. The sheltering facility has sufficient capacity, in the dormitory cells, gyms and chapel, and additional auxiliary spaces to serve as a secure housing unit for the entire CMC population. A contingency plan is in place for operation with a skeleton crew, in scenarios when limited staff would be available.

The FFT visited the CMC East Facility inmate cells that would be used in the event of a sheltering in place event. The facility has centralized ventilation, and both fixed and openable windows. The single-occupancy cells can be as small as 400 ft³. It was not clear for what duration the facility could safely operate with both the mechanical ventilation turned off and all the windows closed. A future conversation with one of the facility managers was proposed to obtain more detailed conversation about ventilation options during a sheltering in place event. The FFT inquired about communications to families of inmates during a sheltering in place event, as to the well-being of the inmates. A follow-up conversation with the Inmate Family Council was proposed.

Sheltering in Place

Unlike the rest of the PAZs, which have either sheltering in place or evacuation as possible Protective Action Measures, correctional facilities only shelter in place. Evacuation Time Estimates (ETE) is are provided in Appendix E of the DCCP Emergency Plan Rev. 5, and updated periodically. The 2022 ETE document states:

“there are three correctional facilities in the EPZ – San Luis Obispo County Jail, San Luis Obispo County Juvenile Hall and California Men’s Colony. The total inmate population at these facilities is 7,135 people. During an emergency all correctional facilities will shelter in place.” (Reference 6.21).

The county’s current guidelines for sheltering in place indicate that they “would likely be short in duration and issued when an evacuation is not feasible,” and that “staying inside with all doors and windows closed and ventilation systems turned off” reduces the chance of inhaling or receiving

body surface contamination from radioactive materials. These guidelines are specific to residential homes: “Lock doors, close windows, air vents and fireplace dampers. Turn off fans, air conditioning and forced air heating systems.” (<https://www.readyslo.org/prepare/actions-in-an-emergency/shelter-in-place/>)

The FFT inquired as to how sheltering in place might be implemented in larger commercial buildings, such as correctional facilities or libraries, in which the duration of safe occupancy while mechanical ventilation is turned off and windows and doors are closed and sealed may be short. For example, if only one hour to several hours are acceptable, in terms of indoor air quality for the spread of communicable diseases, CO₂ levels, temperature, humidity, oxygen, thermal comfort etc., how do the sheltering in place plans integrate building-specific information into development of the county’s decisions for protective actions during a site-area emergency? The drill exercise did not include off-site sheltering in place decisions by the county, so the FFT did not have the opportunity to observe how this might unfold in this particular drill exercise. Future conversations with the county should further explore the questions of sheltering in place.

Conclusions: An emergency operational plan is in place for responding to Radiological Incidents at the California Men’s Colony (CMC). CMC welcomed further integration into county radiological emergency preparedness drills and training.

Recommendations to the DCISC: The DCISC should follow-up with the California Men’s Colony (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 the California Men’s Colony (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)

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 of which 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)

Recommendations to PG&E: None.

3.12 Public Outreach – Interview with Local Radio Station

On the Morning of 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. Ms. Hanna began her interview by asking Dr. Scarlet about her educational background and about her role as a Member of the DCISC. 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 and seeks to learn from the public as well as to provide information that is relevant to the audience.

Dr. Scarlet described how the constituents are selected for the three-year DCISC appointment. Specifically, she stated the three DCISC Members were selected by the California Governor, the State Attorney General, and the Chair of the California Energy Commission. Each of these officials select an individual once every three years. If the candidate has been in service for a while, they might submit a notice of resignation, or they may indicate they would be willing to sign up for another three-year term. Members of the DCISC in the past have generally been in service for several active terms on the Committee.

The discussion turned to the subject of nuclear safety, Ms. Hanna was unfamiliar with the workings of a Nuclear Plant, like DCP, and Dr. Scarlet provided a high-level overview of what these plants do and how they are regulated by both national regulations as well as state and county permitting requirements. Dr. Scarlet described what the technical people at the plant do to ensure safety is always the first measure. She spent a few minutes describing Probabilistic Risk Assessment (PRA), risk-informed decision-making, and how the Nuclear Regulatory Commission (NRC) evaluates the safe operation nuclear power plants.

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 to reach out to others that may have questions regarding DCP and invited attendance to the DCISC public meetings, with the upcoming meeting being in October 2026.

Conclusions: Taking the time to reach out to an interested public stakeholder is an important part of establishing trust and information exchange. 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.

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

4.0 CONCLUSIONS

4.1 DCP 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, and made its submission to the NRC on June 29, 2026, 4 months ahead of the 18-month regulatory schedule. Capsule B results and analysis are publicly available.

- 4.2 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 an attached emery cloth. The final ISFSI inspection report is being prepared by PG&E.
- 4.3 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. The initial indications were that effects are small, and DCPD continued to monitor progress.
- 4.4 In the past, PG&E planned to conduct an "expedited spent fuel transfer" for decommissioning, and this approach was to minimize the overall decommissioning schedule and resulting cost to ratepayers. This expedited spent fuel transfer was not to address a safety concern since PG&E, UCLA, and NRC conclusions all demonstrated that spent fuel can be safely stored in either the SFPs or the ISFSI. PG&E currently plans to conduct offload campaigns on a nominal, non-expedited two-year frequency to maintain available Spent Fuel Pool space for continued DCPD operations. This frequency will be reevaluated if the State approves legislation to operate DCPD beyond 2030.
- 4.5 The meeting with the NRC Senior Resident Inspector was beneficial, and the DCISC should continue the periodic meetings.
- 4.6 The regular meetings with DCPD Officers and Directors continued to be beneficial for both organizations.
- 4.7 DCPD continues to maintain a mature and proactive Flow Accelerated Corrosion Program that remains supported by plant management, engineering organizations, and industry operating experience.
- 4.8 DCPD's update on recent station activities was informative.
- 4.9 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 was not yet available, 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.

- 4.10 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 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.
- 4.11 An emergency operational plan is in place for responding to Radiological Incidents at the California Men's Colony (CMC). CMC welcomed further integration into county radiological emergency preparedness drills and training.
- 4.12 Taking the time to reach out to an interested public stakeholder is an important part of establishing trust and information exchange. 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.

5.0 RECOMMENDATIONS

Recommendations to the DCISC:

- 5.1 The DCISC should issue an explanation for an audience of general public and public officials, written by a panel of three to five experts 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.
 - 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, probabilistic fracture toughness analysis. (D9/26FF-3.1A)

- 5.2 The DCISC should follow-up on quality assurance oversight of vendor-supplied safety-related data and analysis, including results of the ongoing PG&E review of Westinghouse's Quality Assurance Program and of the upcoming NUPIC audit. (D9/26FF-3.1B)
- 5.3 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)
- 5.4 DCISC should routinely review spent fuel transfer campaign planning and this activity should be added to the recurring Open Items List. (D9/26FF3.4)
- 5.5 The following previous recommendation to the DCISC was reviewed in this Fact-Finding Meeting (Section 3.5) 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-3.5)
- 5.6 The DCISC should review the DCCP September 2, 2026, Evaluated Emergency Preparedness Exercise critique and NRC and FEMA evaluations in a future fact-finding meeting when they are completed. (D9/26FF-3.9)
- 5.7 The DCISC should follow-up with San Luis Obispo County to identify the effectiveness with which public communication goals are met, as described in the DCCP 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)
- 5.8 The DCISC should follow-up with the California Men's Colony (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)
- 5.9 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)

- 5.10 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. (D9/26FF-3.11C)
- 5.11 The DCISC should follow-up with SLO County on the integration of emergency preparedness, training, and drills with state and county correctional facilities, all of which 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)
- 5.12 Establish a plan and allocate the necessary resources for DCISC's public engagement scope. (D9/26FF-3.12A)

Recommendations to PG&E:

- 5.10 The following previous recommendation to PG&E was reviewed in this Fact-Finding Meeting (Section 3.1) 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)

6.0 REFERENCES

- 6.1 "Diablo Canyon Independent Safety Committee Thirty-Seventh Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2026 – June 30, 2027," Scheduled for Approved October 21, 2026, DCISC Public Meeting, Volume II, Exhibit D.1, Section 3.7, "Reactor Vessel Embrittlement Sample Capsule B Report."
- 6.2 "Analysis of Capsule B from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-19173-NP, Revision 0, June 2026, submitted to the Nuclear Regulatory Commission as an attachment to PG&E letter DCL-26-046, June 29, 2026, NRC ADAMS Accession Number ML26180A296.
- 6.3 "Diablo Canyon Independent Safety Committee Thirty-Fourth Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2023 – June 30, 2024," Approved October 10, 2024, Executive Summary, Recommendations, "PG&E Recommendation No. 23-2."
- 6.4 "Safety Evaluation Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2," June 2025, U.S. Nuclear Regulatory Commission, NRC ADAMS Accession Number ML25153A508.

- 6.5 “WCAP-18924-NP, Diablo Canyon Units 1 and 2 Initial License Renewal: Time-Limited Aging Analyses (TLAAs) on Reactor Vessel Integrity (RVI),” Revision 0, August 2024, submitted to the Nuclear Regulatory Commission as Enclosure 5 to PG&E letter DCL-24-092, October 14, 2024, NRC ADAMS Accession Number ML24289A118.
- 6.6 “Diablo Canyon Independent Safety Committee Thirty-Sixth Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2025 – June 30, 2026,” Scheduled for approval at the October 21, 2026, DCISC Public Meeting, Volume II, Exhibit D.7, Section 3.12, “Preparation for Future Independent Spent Fuel Storage Installation (ISFSI) Inspections in August 2026.”
- 6.7 “Diablo Canyon Independent Safety Committee Thirty-Sixth Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2025 – June 30, 2026,” Scheduled for approval at the October 21, 2026, DCISC Public Meeting, Volume II, Exhibit D.1, Section 3.11, “Independent Spent Fuel Storage Installation (ISFSI) Update.”
- 6.8 “Diablo Canyon Independent Safety Committee Thirty-Sixth Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2025 – June 30, 2026,” Scheduled for approval at the October 21, 2026, DCISC Public Meeting, Volume II, Exhibit D.3, Section 3.4, “Spent Fuel Canister Corrosion.”
- 6.9 Westinghouse reported a “Fuel Pool Storage Criticality Calculation Error as per “10CFR21 Reporting of Defects and Noncompliance.” Westinghouse submitted a Part 21 to the NRC on June 19, 2026. The NRC classified it as a 10CFR Section: 21/21(a)(2) - Interim Evaluation of Deviation”. The NRC Event Number is 58325, ML26170A012.
- 6.10 Proposed Re-Racking Spent Fuel Pool 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. DCISC has asked DCPD for its position on these two proposals.
- 6.11 “Diablo Canyon Independent Safety Committee Thirty-Seventh Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2026– June 30, 2027” Scheduled for approval at the October 19, 2027, DCISC Public Meeting, Volume II, Exhibit D.2, Section 3.2 “Meet with Nuclear Regulatory Commission Resident Inspector.”
- 6.12 “Diablo Canyon Independent Safety Committee Thirty-Seventh Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2026– June 30, 2027” Scheduled for approval at the October 21, 2026, DCISC Public Meeting, Volume II, Exhibit D.2, Section 3.3 “Meet with a DCPD Officer.”
- 6.13 “Diablo Canyon Independent Safety Committee Thirty-Second Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2021 – June 30, 2022,” Approved October 19, 2022, Volume II, Exhibit D.9, Section 3.7, “Flow Accelerated Corrosion (FAC).”

- 6.14 “Diablo Canyon Independent Safety Committee Thirty-Seventh Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2026– June 30, 2027” Scheduled for approval at the October 19, 2027, DCISC Public Meeting, Volume II, Exhibit D.2, Section 3.7 “Station Update.”
- 6.15 “Diablo Canyon Independent Safety Committee Thirty-Fifth Annual Report on the Safety of Diablo Canyon Nuclear Power Plant Operations, July 1, 2024 – June 30, 2025,” Approved at the October 23, 2025, DCISC Public Meeting, Volume II, Exhibit D.1, Section 3.1, “Observation of the DCPN NRC-Evaluated Emergency Exercise.”
- 6.16 Sellnow, D. D., Lane, D. R., Sellnow, T. L., & Littlefield, R. S. (2017). The IDEA model as a best practice for effective instructional risk and crisis communication. *Communication Studies*, 68, 552-567. <https://doi.org/10.1080/10510974.2017.1375535>
- 6.17 Sellnow T. L., & Sellnow D. D. (2024). *Before crisis: The practice of effective risk communication*. Cognella Academic Publishing.
- 6.18 Kim, H. J., & Lee, R. (2024). Examining radiation emergency preparedness among citizens of Busan, South Korea: Based on the IDEA framework. *International Journal of Disaster Reduction*, 83, 103432. <https://doi.org/10.1016/j.ijdr.2022.103432>
- 6.19 Muhammed T, S., & Mathew, S. K. (2022). The disaster of misinformation: a review of research in social media. *International journal of data science and analytics*, 13(4), 271-285. <https://doi.org/10.1007/s41060-022-00311-6>
- 6.20 Sutton, J., & Wood, M. M. (2025). Opting out: over-alerting and warning fatigue in the era of wireless emergency alerts. *Journal of Contingencies and Crisis Management*, 33(3), e70076. https://scholarsarchive.library.albany.edu/ehc_fac_scholar/20
- 6.21 “Diablo Canyon Power Plant Evacuation Time Estimate Analysis Using Most Recent Decennial Census Data, PG&E Letter DCL-22-071”. “2022 DCPN Evacuation Time Estimate, NRC ADAMS ML22255A196.”

PG&E Presentation Slides from June 30 and July 7, 2026, Briefing on Sample Capsule B Test Results

DCPP Unit 1 Capsule B Testing and Analysis

Results Briefing to DCISC

June 30 - July 7, 2026



Together, Building
a Better California



Summary

- Westinghouse performed extensive testing on Surveillance Capsule B per 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 documented in non-proprietary report WCAP-19173-NP and submitted to NRC on June 29, 2026

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PG&E Presentation Slides from June 30 and July 7, 2026, Briefing on Sample Capsule B Test Results (continued)



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:
 - 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 54 EFPY, unirradiated USE)

EFPY = Effective Full Power Years

Public

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Credibility Evaluation

- 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)

Public

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PG&E Presentation Slides from June 30 and July 7, 2026, Briefing on Sample Capsule B Test Results (continued)



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 54 EFPY (60 years):**
same limiting locations as previous analysis

Public

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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

Public

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Communication between Dr. Frank N. von Hippel and the DCISC

The following communication was sent via email to the Diablo Canyon Independent Safety Commission (DCISC) on Friday, June 5, 2026 by Dr. Frank N. von Hippel. Dr. Hippel is a Senior Research Physicist and Professor of Public and International Affairs emeritus. He also is involved in the Program on Science and Global Security at Princeton University. Dr. Hippel's email address is fvhippel@princeton.edu.

The DCISC has requested Diablo Canyons' position on Dr. Hippel's recommendation for a proposed expedited transfer of spent fuel to the ISFSI and re-racking the spent fuel pools.

Dear members of the Diablo Canyon Independent Safety Commission:

During 2012-2016, I served on the Congressionally-mandated National Academies (NAS) panel on Lessons Learned from the Fukushima Nuclear Accident. Professor Meshkati served as an advisor to the study committee for first phase of our study.

A major focus of our work was on a spent fuel pool fire that almost happened in Fukushima spent-fuel pool #4. Indeed, for a period of about two weeks or so, it was believed that the spent fuel in the pool was on fire. The reason was a hydrogen explosion on March 15, 2011 that destroyed the roof and most of the walls of the structure over that pool. The reactor core of unit 4 had been moved into the pool so that the pressure vessel could be serviced. Any hydrogen generating steam-zircalloy reactions in unit 4 would therefore have had to have occurred in the pool. Japan's Prime Minister was informed that, if the wind blew toward Tokyo, 100 miles to the south, the city might have to be evacuated (see figure 5 in "Reducing the Danger from Fires in Spent Fuel Pools" attached).

Later, when it became clear that the spent fuel in pool 4 was still covered with water, it was concluded that the hydrogen that blew up the building over pool #4 must have leaked into the building via a venting system shared with unit 3.

However, the NAS study group devoted one of the two volumes of our report to the danger of spent fuel pool fires, <https://www.nationalacademies.org/projects/DELS-NRSB-12-01/publication/21874>.

In parallel, the Nuclear Regulatory Commission did its own "lessons learned" studies, including a "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," NRC COMSECY-13-0030, 2013, <https://www.nrc.gov/docs/ML1327/ML13273A628.pdf>. "Expedited transfer" of spent fuel to dry-cask storage after five years of cooling in the pool would make possible a return to open racking in the pool and, in the NRC staff's view, would have eliminated the chance of a hydrogen explosion destroying the building covering the pool and therefore would have greatly

Communication between Dr. Frank N. von Hippel and the DCISC (continued)

reduced the release of cesium-137 into the human environment and thereby the need to evacuate a large area.

The NRC study estimated the costs to US utilities and benefits to the US public of a possible regulatory requirement to return US spent fuel storage pools to their original “open-rack” designs in which the spent fuel assemblies were separated enough so that no criticality will not occur and in which air cooling would be possible if there were a loss of water whereas, in their current dense-packed design, spent fuel assemblies are stored almost as densely as in reactor cores with each fuel assembly in a box of borated steel to prevent criticality,.

The staff found that, for dense racking, a fire following the uncovering of the spent fuel in an average US spent fuel pool would result in the release to the atmosphere of two thirds of the 30-year half-life Cesium-137 in the stored spent fuel – about 100 times as much as leaked out of the containments of Fukushima units 1,2 and 3 after their core meltdowns in 2011. Cesium-137 contamination determined the areas of long-term population relocation following the Chernobyl and Fukushima accidents.

In contrast, the NRC staff found that, with a return to open racking, if the water drained out of a spent fuel pool, not enough hydrogen would be produced to destroy the building covering a pool and that, as a result, even if there were a spent fuel fire, only about 1.5% as much cesium-137 would be released into the atmosphere.

Returning to open racking would require the utilities to purchase dry cask storage earlier. The staff estimated that delaying purchases of the casks cumulatively earned the utilities about \$50 million in interest on the funds per pool.

The NRC staff calculated the benefits to the public of returning to open rack storage as the probability-weighted reduction of economic losses from the smaller population relocated. With regard to probability, they assumed that the only possible cause of a loss of water would be by a super earthquake to which they assigned an average probability of 4.3×10^{-6} per pool year. (Obviously, seismically, Diablo Canyon is not an average site.) When I asked about the probability of a successful terrorist attack on a spent fuel pool, the staffer briefing us said that the NRC did not know how to calculate that probability and therefore had assumed it was zero. They also did not calculate the probability of a different scenario for Fukushima pool #4. There, because of the high radiation level on the pool deck, it was filled blind, using a pump with a boom designed for pumping cement up several floors during the construction of buildings. Unbeknownst to the operators most of the water missed the pool and the spent fuel only remained covered because of fortuitous leakage of water into the pool through the gate blocking the canal to the filled reactor cavity (see Figure 2.15 of the National Academy report).

We didn't know how to calculate the probability of a spent-fuel-pool fire either, so we focused our questions on the NRC's staff's consequence calculations, which we *did* know how to calculate. Unfortunately, the NRC staff did not respond to any of our questions about their assumptions until after our report was submitted to Congress.

Communication between Dr. Frank N. von Hippel and the DCISC (continued)

Even then, the NRC staff responded to only one of our questions. We asked the contamination threshold level used for population relocation. We learned that they had assumed a threshold Cs-137 contamination for population relocation three times higher than that used for the Chernobyl and Fukushima accidents.

At the same time, the State of New York, which was reviewing calculations the NRC staff had done for the consequences of a hypothetical reactor accident at Indian Point, up the Hudson River from New York City, asked the NRC about its assumption that it would be possible to decontaminate land by a factor of 15 within a year and thereby limit the relocation period for almost all the population to less than a year. That assumption differed from Japan's experience at Fukushima where the largest decontamination factor achieved after several years was by a factor of three. The New York State officials asked where the NRC got its assumption. The NRC searched but had to report back that it could not determine the basis for that assumption.

When a colleague and I redid the NRC's calculations using the Chernobyl and Fukushima contamination evacuation thresholds and assuming a decontamination factor of 3 after two years, we found economic damages from a fire in a dense-packed spent-fuel pool at an average US site of about \$2 trillion, 16 times the NRC staff's estimate of \$0.125 trillion. Multiplying by the staff's estimate of a probability of 4.3×10^{-6} per pool year times an estimated discounted average remaining pool life of 10.7 years resulted in a corrected probability-weighted benefit from expedited transfer of \$90 million per pool vs. the \$7 million the NRC calculated. Therefore, correcting just those two errors without considering the probability of other scenarios for pool drainage would have reversed the conclusion of the NRC's calculation that the costs to the utilities of returning to open pool racking would exceed the probability-weighted losses to the public. More details may be found in the attached article, "Economic Losses from a Fire in a Dense-Packed U.S. Spent Fuel Pool."

<https://scienceandglobalsecurity.org/archive/sgs25vonhippel.pdf>

How is this relevant to Diablo Canyon?

The report on "Spent Fuel Management" by the Diablo Decommissioning Engagement Panel, <https://diablocanyonpanel.org/decom-topics/spent-fuel-management/> shows that the pools are dense racked to five times the original density with capacity for 30 years discharges). The pools are not completely full, but the assemblies are in individual cells with solid borated walls to prevent criticality, which would prevent air cooling until the water in the pool drained completely, at which point some air could enter the holes at the bottoms of the racks. This would most likely result in a spent fuel fire if the fuel were partially uncovered.

Communication between Dr. Frank N. von Hippel and the DCISC (continued)

The Nuclear Regulatory Commission's "Diablo Canyon Fact Sheet: Seismic and Tsunami Issues," <https://www.nrc.gov/docs/ml1112/ml111290158.pdf>, states that: "The base of the pools are below grade level, on bedrock, and the walls of lower portions of the pools are also below grade, such that the top of the spent fuel storage racks are near the exterior ground level to the east of the Auxiliary Building."

Since the plant is built on a slope downward to the ocean to its west, that probably means that the tops of the racks are above ground level on the west. A severe earthquake or terrorist attack could therefore drain the pool to a level that could result in a spent fuel fire. Steam-zirconium reactions would generate hydrogen, and a hydrogen explosion could destroy the building above the spent fuel pools as occurred at Fukushima.

A cost-benefit analysis could be done for Diablo Canyon, taking into account the wind pattern and surrounding population density. Given the uncertainty in the probability of a loss of water, however, and that the pools were far from full – at least in 2018, according the Decommissioning Engagement Panel, however, I think that your committee would be justified in recommending that the Diablo Canyon pools be returned to open racking. Given the fact that the pools are not full, the reracking and early purchase of the dry casks might cost considerably less than \$100 million.

I therefore think that your Diablo Canyon Independent Safety Committee would be justified in recommending that the Diablo Canyon reactors' spent fuel pools be returned to open racking and that any spent fuel older than 5 years be transferred to dry cask storage. Based on the power generated by the two reactors in

2024, <https://pris.iaea.org/PRIS/CountryStatistics/CountryDetails.aspx?current=US>, even a cost of \$100 million would be paid back in a year by a surcharge of 0.5 cents/kWh.

Given the potentially catastrophic consequences of a spent fuel pool fire, I think that the ratepayers would likely consider that a bargain and other states might well pay attention.

Sincerely yours.

Frank N. von Hippel
Senior Research Physicist and Professor of Public and International Affairs emeritus
Program on Science and Global Security, Princeton University

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Communication between the DCISC and PG&E

DCISC Request:

Dr. Frank von Hippel has written a letter to DCISC raising the concern of fires in spent fuel pools and proposing expedited spent fuel transfer to dry casks in order to return spent fuel pools to "open racking" spacing.

DCISC is requesting PG&E's position on the proposed expedited transfer of spent fuel to the ISFSI.

Dr. Frank von Hippel's Recommendation:

"Since the plant is built on a slope downward to the ocean to its west, that probably means that the tops of the racks are above ground level on the west. A severe earthquake or terrorist attack could therefore drain the pool to a level that could result in a spent fuel fire. Steam-zirconium reactions would generate hydrogen, and a hydrogen explosion could destroy the building above the spent fuel pools as occurred at Fukushima.

A cost-benefit analysis could be done for Diablo Canyon, taking into account the wind pattern and surrounding population density. Given the uncertainty in the probability of a loss of water, however, and that the pools were far from full – at least in 2018, according the Decommissioning Engagement Panel, however, I think that your committee would be justified in recommending that the Diablo Canyon pools be returned to open racking. Given the fact that the pools are not full, the reracking and early purchase of the dry casks might cost considerably less than \$100 million.

I therefore think that your Diablo Canyon Independent Safety Committee would be justified in recommending that the Diablo Canyon reactors' spent fuel pools be returned to open racking and that any spent fuel older than 5 years be transferred to dry cask storage."

PG&E Response:

Summary

During DCPD Decommissioning Planning in 2018-2022, PG&E conducted detailed reviews of the holistic spent fuel storage strategy, offload timeframes from the spent fuel pools (SFPs) to the onsite Diablo Canyon Independent Spent Fuel Storage Installation (DC ISFSI), and associated risk. Consistent with NRC analysis conclusions, PG&E's current position remains that spent fuel is safely stored in both the SFPs and at the DC ISFSI. PG&E currently plans to conduct spent fuel transfers to the DC ISFSI (via "offload campaigns") on a nominal 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.

Based on the existing SFPs design, operational procedures, previously-conducted PG&E and UCLA evaluations, and NRC conclusions, PG&E has no plans to return the SFPs to "open racking" nor transfer spent fuel to the DC ISFSI only because it has been stored for greater than five years.

Communication between the DCISC and PG&E (continued)

UCLA Study: "A Probabilistic Risk Assessment of Nuclear Power Plant Spent Fuel Handling and Storage Programs: Methodology and Application to the Diablo Canyon Power Plant"

A first-of-a-kind risk assessment was performed by the B. John Garrick Institute for the Risk Sciences at UCLA to compare the degree of radiological safety to the public associated with storing DCPD spent fuel in the SFPs and at the DC ISFSI to determine which spent fuel transfer approach has the lowest risk. This risk assessment considered four spent fuel transfer options, each of which had different timetables for emptying the SFPs, a different number of spent fuel assemblies in the SFPs at a given time, and a different overall heat load in the SFPs at a given time. Risks associated with all four spent fuel transfer options were found to be extremely small (less than 1% of the NRC's objectives for safe operation of nuclear plants).

In the 30th Annual Report on the Safety of Diablo Canyon Nuclear Power Plant for the Period July 1, 2019 -June 30, 2020,¹ DCISC concluded and recommended, in part:

"The DCPD-sanctioned spent fuel risk assessment performed by The B. John Garrick Institute for the Risk Sciences at UCLA appeared well-developed and focused. The assessment found small differences in risk among the four options analyzed, and all were within the NRC's spent fuel storage risk limits. The smallest risk was for the option of early movement of spent fuel from the DCPD Spent Fuel Pool to the Independent Spent Fuel Storage Installation beginning following the Unit 1 shutdown and prior to the Unit 2 shutdown.

...

Recommendation: The DCISC recommends that when PG&E considers decisions about the future management on-site of the spent fuel from DCPD's two reactor units, the risks arising from spent fuel management should be one part of the PG&E decision process and that process should be informed by the conclusions contained in the Study entitled A Probabilistic Risk Assessment of Nuclear Power Plant Spent Fuel Handling and Storage Programs: Methodology and Application to the Diablo Canyon Power Plant."

In alignment with this recommendation, PG&E has considered the risk insights from the UCLA Study during decision-making such that structured offload campaigns are planned to maintain available SFP space for continued DCPD operations.

Existing SFP Design and Procedures

As discussed in the UCLA Study, the extensive SFP design provides defense against severe SFP accidents, such as loss of SFP water.

- The DCPD SFPs are robust structures, each consisting of 6-foot thick walls of reinforced concrete lined with stainless steel and the structure base in bedrock.
- SFP temperature and water levels are continuously monitored, with annunciation in the Control Room if setpoints are reached. In addition, area radiation monitors provide continuous radiation levels for the SFP and new fuel storage areas.

¹ [Thirtieth Annual Report on the Safety of the Diablo Canyon Nuclear Power Plant, by the Diablo Canyon Independent Safety Committee](#), Executive Summary Conclusion #20 and Recommendation.

Communication between the DCISC and PG&E (continued)

- There are multiple ways to initiate mitigative actions and add makeup water to the SFPs with or without entry to the SFP floor. Each SFP has a completely independent SFP cooling system. Backup SFP makeup capability is also available using pumps and gravity feed from the raw water storage reservoirs that can supply makeup water to the SFPs via a separate system. PG&E also has an onsite fire engine (i.e., portable diesel-driven pump) capable of taking suction from various water sources onsite through portable flexible hoses and providing makeup.

Furthermore, PG&E maintains procedures and conducts extensive training to respond to a potential SFP accident. These mitigative strategies are maintained in accordance with License Condition 2.1 of the DCPD Units 1 and 2 Facility Licenses.

These design features and diverse strategies provide defense-in-depth and ample time to provide makeup water or spray to the SFPs prior to the onset of zirconium cladding ignition (i.e., a steam-zirconium reaction), thus, preventing the generation of hydrogen and release of radioactivity to the environment.

Analysis of SFP Zirconium Fire

Dr. von Hippel hypothesizes potential hydrogen generation and resulting explosion from a steam-zirconium reaction (zirconium fire).

The risk associated with zirconium cladding fire events decreases as the spent fuel ages, decay time increases, decay heat decreases, and short-lived radionuclides decay away. As decay time increases, the overall risk of a zirconium cladding fire continues to decrease due to two factors: (1) the amount of time available for preventative actions increases, which reduces the probability that the actions would not be successful; and (2) the increased likelihood that the fuel is able to be cooled by air, which decreases the reliance on actions to prevent a zirconium fire.

To support DCPD Decommissioning planning efforts, an adiabatic heatup analysis was performed that conservatively evaluated how many hours it takes for uncovered spent fuel assemblies in the DCPD SFPs to reach the temperature at which the zirconium cladding would fail (900°C). Per ²,² NRC has determined that 10 hours is acceptable to initiate mitigative actions or, if needed, for offsite authorities to implement protective actions using a comprehensive emergency management plan approach.

² NSIR/DPR-ISG-02, Revision 0, "Interim Staff Guidance, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," dated May 2015 ([ML14106A057](#)).

In the DCPD adiabatic heatup analysis, the SFP draining process was assumed to be instantaneous, which is conservative since no credit is given to partial cooling by SFP water as the draining process occurs. The analysis concluded that it would take almost 29 hours for an uncovered spent fuel assembly to reach 900°C after five years of storage (the time at which Dr. von Hippel recommends transfer from the SFPs to the DC ISFSI). Given NRC finds 10 hours acceptable, this demonstrates DCPD has more than sufficient time to mitigate events that could lead to a zirconium cladding fire, thus, preventing the generation of hydrogen and release of radioactivity to the environment.

Communication between the DCISC and PG&E (continued)

Analysis of Radiological Impact from a Complete Loss of SFP Water

Dr. von Hippel hypothesizes potential radiological impact to local communities from a complete loss of SFP water (drain down).

The NRC evaluates radiological consequences, in part, by following the Environmental Protection Agency (EPA) Protective Action Guides (PAGs) manual. This manual contains guidelines that would trigger public safety measures during a radiation leak or accident that created a mobile airborne plume.

To support DCPD Decommissioning planning efforts, an analysis was conducted to determine the radiological impact of a complete loss of SFP water by comparison to EPA PAGs at the DCPD exclusion area boundary (EAB). The analysis conservatively assumed that the SFP racks were completely full. It was determined that the gamma radiation dose rate at the EAB would be less than the EPA PAGs exposure levels after five years of storage (the time at which Dr. von Hippel recommends transfer from the SFPs to the DC ISFSI). Therefore, Dr. von Hippel's hypothesized impact to local communities is not applicable given DCPD's SFP design, isolated location, and spent fuel inventory.

SFP Seismic Assessment

Dr. von Hippel has noted that DCPD is not an average site when it comes to seismicity and that a severe earthquake could drain the SFPs.

In response to an NRC request for information in 2012 after the Fukushima Dai-ichi accident, PG&E performed a site-specific seismic assessment including a seismic hazard re-evaluation of the site and an evaluation of SFP seismic integrity. The SFP seismic evaluation was prepared and submitted³ for NRC review consistent with NRC-endorsed⁴ EPRI "Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation" (SFP Evaluation Guidance Report).⁵

³ PG&E Letter DCL-17-108, "Spent Fuel Pool Evaluation Supplemental Report. Response to NRC Request for Information Pursuant to 10 CFR 50.54(0) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated December 18, 2017 ([ML17352A703](#)).

⁴ NRC Letter, "Endorsement of Electric Power Research Institute Report 3002009564, 'Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation,'" dated February 28, 2017 ([ML17034A408](#)).

⁵ EPRI Technical Report No. 3002009564, "Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation," dated January 2017 ([ML17031A176](#)).

⁶ NRC Letter "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Staff Review of Spent Fuel Pool Evaluation Associated With Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1: Seismic," dated August 8, 2018, ([ML18211A322](#)).

Communication between the DCISC and PG&E (continued)

The NRC conclude⁶⁷ that the SFP Evaluation Guidance Report methodology was followed when calculating the site-specific seismic capacity of the SFPs, and that DCP's site-specific values and conditions are within the acceptable limits and bounds considered for the non-structural evaluation criteria specified in the SFP Evaluation Guidance Report. In particular, NRC confirmed that there is high confidence that the SFP structure is sufficiently robust to withstand ground motions with peak spectral accelerations up to and including the peak spectral acceleration of DCP's ground motion response spectra. The NRC also concluded that PG&E adequately evaluated the non-structural considerations for components whose failure could lead to potential drain-down of the SFP due to a seismic event. Further, the NRC concluded that PG&E demonstrated a potential drain-down of the SFP as a result of the reevaluated seismic hazard is unlikely.

Conclusion

In 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 DC ISFSI, PG&E currently plans to conduct offload campaigns on a nominal, non-expedited two-year frequency to maintain available SFP space for continued DCP operations. This frequency will be reevaluated if the State approves legislation to operate DCP past 2030.