GIRS-2020-3/L

The B. John Garrick Institute for the Risk Sciences
UCLA ENGINEERING

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

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Prepared Under
A Collaborative Research and Development Agreement Between
The B. John Garrick Institute for the Risk Sciences at UCLA
and
Pacific Gas and Electric Company

February 17, 2020

ACKNOWLEDGMENTS

This This research and development project required the collaboration of many professionals at both PG&E and The B. John Garrick Institute at UCLA. The authors wish to thank PG&E for the professional manner in which they handled all requests for information. The efficiency with which documents were provided made possible performing an extensive amount of analysis in a short period of time. The PG&E staff were exemplary in their support of the study.

Special thanks to John Kessler, Consultant, who provided critically needed information and review and Mihai A. Diaconeasa of the Garrick Institute for providing technical input for planning and scoping the study. We also wish to thank Ali Mosleh, Director of the Garrick Institute, for his leadership in collaborating with PG&E. We wish to acknowledge and thank Maureen Zawalick and Jordan Tyman in establishing and monitoring the collaborative research program between the UCLA Garrick Institute and PG&E under which this study was conducted. Finally, the authors want to thank the Garrick Institute staff for their assistance in publishing the study.

ACRONYMS AND ABBREVIATIONS

AB	Auxiliary Building
ACRS	Advisory Committee on Reactor Safeguards
AFW	Auxiliary Feedwater
ASW	Auxiliary Saltwater
BDB	Beyond Design Basis
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CCW	Component Cooling Water
CFD	Computational Fluid Dynamics
CTF	Cask Transfer Facility
CST	Condensate Storage Tank
CWA	Cask Washdown Area
DCPP	Diablo Canyon Power Plant
DDE	Double Design Earthquake
DFC	Damaged Fuel Container
DSFR	DCPP Spent Fuel Risk
EDG	Emergency Diesel Generator
EOL	End of Life
EPRI	Electric Power Research Institute
FDF	Fuel Damage Frequency
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
FWST	Fire Water Storage Tank
HCLPF	High Confidence (85%) of Low Probability of Failure (<.05)
HEP	Human Error Probability
HEPA	High-Efficiency Particulate Air (filter)
HRA	Human Reliability Analysis
ILP	ISFSI Long Period (Earthquake Spectra)
ISFSI	Independent Spent Fuel Storage Installation
K_{eff}	Effective Reactivity Coefficient, k-effective
LERF	Large Early Release Frequency

LPT	Low Profile Transporter
LTSP	Long Term Seismic Program
MLWL	Mean Lower Water Level
MPC	Multipurpose Canister
NPP	Nuclear Power Plant
PG&E	Pacific Gas and Electric
ppm	Parts Per Million
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objectives
RHR	Residual Heat Removal
RWR	Raw Water Reservoir
RWST	Refueling Water Storage Tank
SA	Spectral Acceleration
SCS	Supplemental Cooling System (for transfer cask)
SFP	Spent Fuel Pool
SNF	Spent Nuclear Fuel
TS	Technical Specifications
UFSAR	Diablo Canyon Spent Fuel Storage Installation "Update for the Final Safety Analysis Report"
USNRC	U.S. Nuclear Regulatory Commission
VCT	Vertical Cask Transporter

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Executive Summary

This study developed a methodology for Probabilistic Risk Assessment of nuclear power plant spent fuel handling and storage programs and demonstrate its application by assessing the radiological risks associated with storage and movement of spent fuel from the spent fuel pools (SFPs) to the Independent Spent Fuel Storage Installation (ISFSI) at the Diablo Canyon Power Plant (DCPP). Both storage methods are regulated by the Nuclear Regulatory Commission (NRC) through licensing, inspection, and enforcement of its requirements.

The DCPP SFPs and DC ISFSI are designed for the site-specific location and conditions, including seismic considerations. Back-up systems and controls are in place to ensure accident consequences are minimized or eliminated. Based on years of analysis and operating experience, the "NRC believes spent fuel pools and dry casks both provide adequate protection of the public health and safety and the environment. Therefore, there is no pressing safety or security reason to mandate earlier transfer of fuel from pool to cask."[1] To put this in a risk-informed perspective, a first-of-a-kind risk assessment was performed to compare the degree of radiological safety to the public associated with storing DCPP SNF in the SFPs and at the DC ISFSI to determine which SNF transfer approach has the lowest risk. Severe earthquakes were determined to be the most-likely cause of accidents (on the order of once in 57,000 years) that could impact DCPP SNF storage.

The risk assessment evaluated four options of transferring the SNF from the SFPs to the DC ISFSI to determine which has the lowest risk.

1. Post-Shutdown 7-Year Offload: Beginning almost 6 years after Unit 2 shutdown, a single offload campaign is performed to complete emptying the SFPs by 7 years after Unit 2 shutdown.
2. Pre-Shutdown 7-Year Offload: This offload scenario includes two offload campaigns conducted prior to Unit 2 shutdown and a third campaign which is to be completed 7 years after the Unit 2 shutdown.
3. Pre-Shutdown 5-Year Offload: This offload scenario includes two offload campaigns prior to Unit 2 shutdown and a third campaign which is to be completed 5 years after the Unit 2 shutdown.
4. Pre-Shutdown Vendor Option Offload: This offload scenario moves many more fuel assemblies than the other pre-shutdown offload scenarios prior to the Unit 2 shutdown. This scenario does not result in completely emptying the SFPs until the end of 2033, more than 1 year later than any of the others.

The risk of each of the four SNF transfer options was found to be very small; considerably less than the USNRC's risk criteria for safe operation. The risk study found that although all options have a low probability of an event occurring and are relatively comparative, considering the inherent uncertainties, the Pre-Shutdown Vendor Offload Option provided the lowest risk of a radiological release event.

Previously-developed generic studies conducted throughout the nuclear industry were reviewed to inform the results for DCPP. As with the studies reviewed there are uncertainties regarding the assumptions and analysis simplifications that could impact the results. Most of the uncertainties impact each transfer option equally and so would not affect the transfer option risk rankings; e.g., uncertainty in the frequency of large seismic events well beyond the design basis. Other uncertainties more directly affect the differences in risk between SNF transfer options. An example of the latter

[1] https://www.nrc.gov/waste/spent-fuel-storage.html.

type of uncertainty is the total SFP decay heat level below which simple natural convection cooling would prevent the SNF from overheating. Thus, additional analysis specific to the DCPP SFPs and DC ISFSI would reduce the uncertainty. For example, human actions are required to respond to an accident (e.g., manual valve operation); however, it is difficult to credit manual human actions during beyond design basis accidents such as a large earthquake due to the lack of human reliability analyses.

In spite of these uncertainties, in context with regulatory risk metrics, i.e., quantitative health objectives, the safety margins of the offload scenarios are considered more than adequate.

Since the study has shown that there are only small differences in risk between SNF transfer options, it is not believed that any significant benefit would be gained from further assessment.

SNF Risk Assessment Results Summary

This risk assessment considered four options of transferring the SNF from the SFPs to the DC ISFSI to determine which has the lowest risk. Each of the options had different timetables for emptying the SFPs, a different number of SNF assemblies in the SFPs at a given time, and a different overall heat load in the SFPs at a given time. The options were as follows:

1. Post-Shutdown 7-Year Offload: Beginning almost 6 years after Unit 2 shutdown, a single offload campaign is performed to complete emptying the SFPs by 7 years after Unit 2 shutdown.
2. Pre-Shutdown 7-Year Offload: This offload scenario includes two offload campaigns conducted prior to Unit 2 shutdown and a third campaign which is to be completed 7 years after the Unit 2 shutdown.
3. Pre-Shutdown 5-Year Offload: This offload scenario includes two offload campaigns prior to Unit 2 shutdown and a third campaign which is to be completed 5 years after the Unit 2 shutdown.
4. Pre-Shutdown Vendor Option Offload**:** This offload scenario moves many more fuel assemblies than the other pre-shutdown offload scenarios prior to the Unit 2 shutdown. This scenario does not result in completely emptying the SFPs until the end of 2033, more than 1 year later than any of the others.

The risk assessment did not evaluate the dose consequences of the options. Rather, it determined the relative risk based on the probability of initiating events. Specifically, the analysis does not calculate offsite radiation doses to the public as a result of a release from a SNF accident. The risk assessment does assess the risk of events (accidents) of different magnitudes and is able to relate the events to regulatory safety metrics and provide evidence of the safety of spent fuel handling operations.

Given the above uncertainty, preliminary conclusions provide five key areas of insight on SNF storage risk. Each insight is briefly summarized below.

1. **Risk Results:**
   - Risks associated with all four SNF transfer options were found to be extremely small (less than 1% of the USNRC's objectives for safe operation of nuclear plants). Controls and programs implemented at DCPP, along with defense in depth of the SFP system design significantly reduce the likelihood of a SNF event.
   - As shown in Figures ES-1 and ES-2, the difference in risk between each of the four SNF transfer options is small compared to the USNRC's objectives for safe operation; however, the assessment found that the uncertainty in the assessment is significant. The risk assessment also found that although all options have a low potential probability of an event occurring and

are relatively comparative, the Pre-Shutdown Vendor Offload Option provided a lower risk of a SNF event.
- Further analysis and model refinements would be needed to reduce uncertainty and to characterize the actual consequences to radiological release.
- In the opinion of the authors of the report, little benefit would be gained from further assessment (i.e., little potential for change in option ranking).

2. **Risk Drivers:**
- Loss of cooling water to the SNF assemblies is the dominant event that leads to radiological release to the Fuel Handling Building environment.
- A key safeguard to overheating is the availability of back-up SFP cooling water. At DCPP, to ensure back-up water is available, water can rapidly be supplied from multiple qualified sources that can cool the SNF for days.
- Loss of SFP cooling water could be caused by the following accidents:
    - A heavy load drop that damages the SFP liner and results in water leakage
    - An earthquake that damages the SFP liner or cooling water components (e.g., pumps, piping, chillers) resulting in water leakage or water heat-up
- Risk of radiological release is dependent on the number of SNF assemblies that overheat. As shown in Figure ES-2, it is more likely to have just a few assemblies overheat versus all assemblies in the SFP. This is because all fuel assemblies in the SFP would be vulnerable to overheating only following a recent offload from the reactor core when the total SFP decay heat is still relatively high. When more than three reactor cores of fuel assemblies are vulnerable to overheating, the oldest core equivalent of fuel assemblies would have already cooled for more than 7 years in the SFP.

Figure ES-1

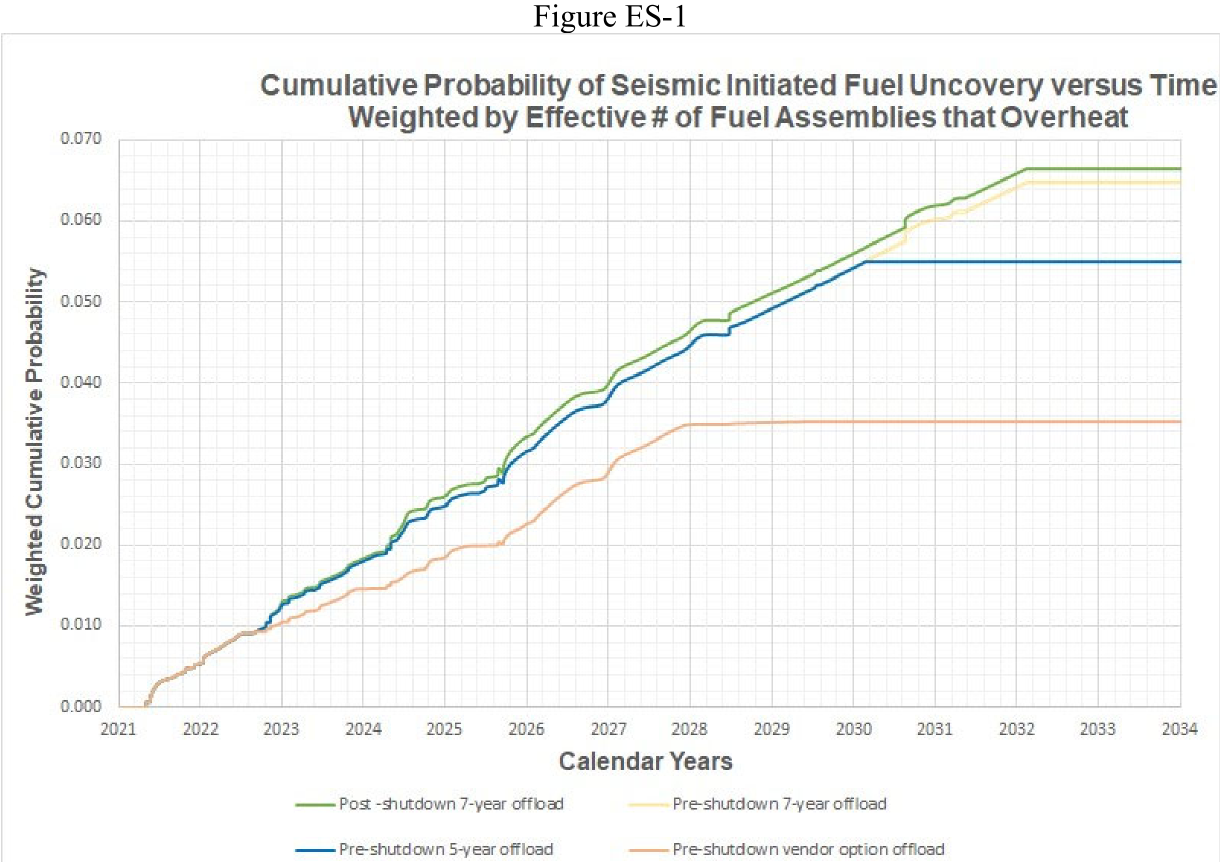
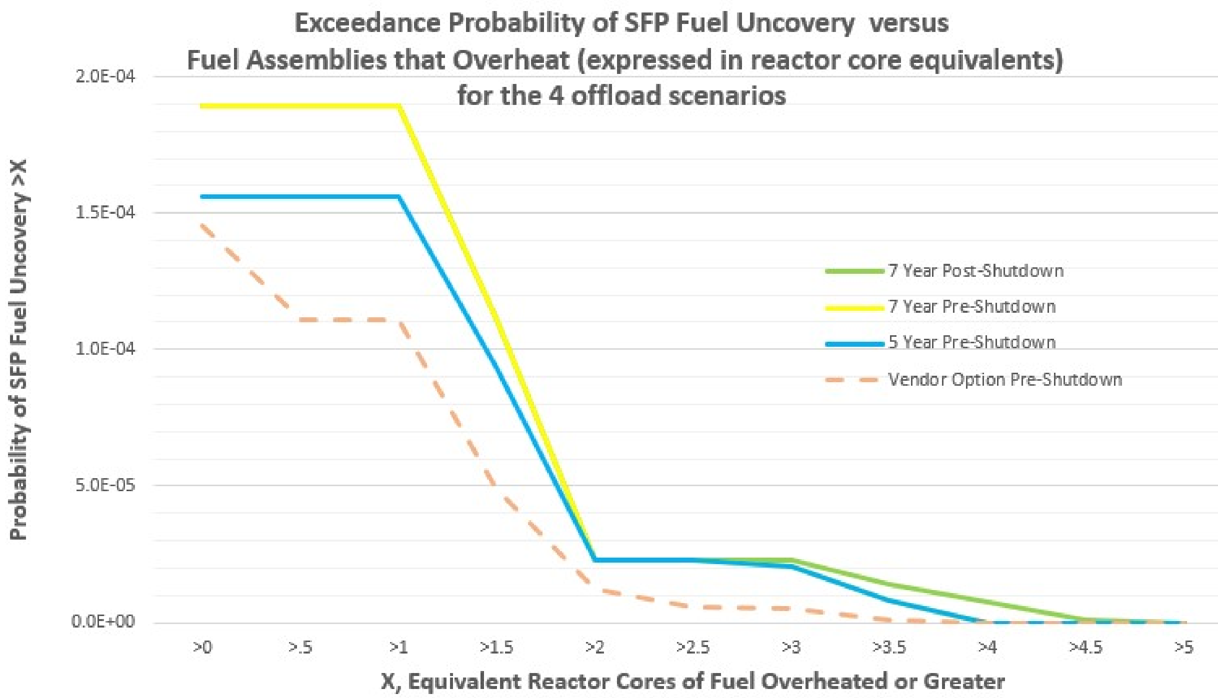


Figure ES-2



3. **Assessment Limitations:**
   - While the assessment was able to rely on previously-developed generic studies and DCPP-specific calculations, several areas of technical limitations were encountered, including:
     - Beyond design basis seismic capacities of some structures and components
     - Assessment of the dose release and effects to the public for a beyond design basis event
     - Evaluation of the SFP cooling hydraulics for potential air-cooling by natural convection within the SFP following a loss of SFP cooling water
     - Analysis of beyond design basis earthquakes affecting mitigating operator actions
   - Significant uncertainties regarding the assumptions and analysis simplifications still remain that could impact the individual option risk results, but not the overall ranking. Thus, additional analysis specific to the DCPP SFPs and DC ISFSI would be needed to reduce the uncertainty.

4. **Uncertainty:**
   - Due to the lack of technical information for the specific scenarios being evaluated, there are several important sources of uncertainty in the assessment:
     - Seismic hazard curve uncertainties – this is not expected to have an impact on the option rankings
     - Beyond design basis seismic capacities of key structures
     - Human error probabilities
     - Heavy load drop frequency – this is not expected to have an impact on the option rankings
     - Extent of SNF overheating, given an SFP uncovery event
   - Some uncertainties have the potential of impacting the relative risks. For example, if the extent of SNF assembly overheating is smaller than currently assumed in the assessment, the differences between the four offload scenarios would be minimized.

- The uncertainties are addressed by bounding assumptions and correlations with DCPP seismic risk studies and long term rigorous industry-wide risk assessments, including those of the USNRC.

5. Recommendation for Next Steps:

- As discussed above, there are technical limitations of the assessment resulting in large uncertainties. Some of the uncertainties impact each transfer option equally and would not affect the transfer option risk rankings; e.g., uncertainty in the frequency of large seismic events well beyond the design basis. Other uncertainties more directly affect the differences in risk between SNF transfer options. An example of the latter type of uncertainty is the total SFP decay heat level below which simple natural convection cooling would prevent the SNF from overheating.
- Additional analysis and potential research would be required to refine the assessment to reduce the uncertainty and provide higher confidence in the results. This would include:
  - Human reliability analysis of post-accident manual actions
  - Determination of beyond design basis seismic capabilities of key structures
  - Heavy load drop frequency determination based on single-failure-proof crane event records
  - Analysis of SFP cooling hydraulics and air cooling thermal analyses
  - Onsite and offsite dose analysis model to determine the doses to workers and the public given use of fuel handling building filtration systems
- As discussed above, since there are only small differences in risk between SNF transfer options, it is not believed that any significant benefit would be gained from further assessment.

Summary

Project Objectives

The purpose of this study to devlop a methodology for Probabilistic Risk Assessment of nuclear power plant spent fuel handling and storage programs and demonstrate its application and by assessing the radiological risks associated with storage and movement spent fuel from the spent fuel pools (SFP) to the Independent Spent Fuel Storage Installation (ISFSI) at the Diablo Canyon Power Plant (DCPP). Pacific Gas & Electric (PG&E) has performed extensive studies of different offload scenarios of the spent fuel from the DCPP SFPs to the ISFSI dry cask storage site (PG&E, 2018a). Four spent fuel offload scenarios have been identified, each having a very high likelihood of being implemented without undue risk to workers and the public. A goal of this study to add transparency to the degree of safety involved to enable making the best decision on the offload scenario having the least risk of radiological consequences to the public.

Description of the Offload Scenarios and Process

The following abstractions of the four spent fuel offload scenarios that resulted from previous studies (Holtec, 2017; TN Americas, 2019) are the basis of this study. These offload scenarios were developed for Unit 2 and are expected to be similar for Unit 1. All four offload scenarios assume Unit 2 final shutdown in late August of 2025.

1. ***Post-Shutdown 7-Year Offload.*** *No offloads to dry storage are performed while the Unit 2 reactor is operating. Beginning in late 2031, almost 6 years after Unit 2 shutdown, a single offload campaign is performed to empty the spent fuel pool by mid-2032. The 7 years is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*

2. ***Pre-Shutdown 7-Year Offload.*** *This offload scenario includes two offload campaigns to dry storage conducted prior to Unit 2 shutdown. The pre-shutdown campaigns are to occur after each of the last two partial core refueling periods, the first of which occurs in 2021. Beginning in early 2032, almost 7 years after Unit 2 shutdown, a third and final offload campaign is then performed to achieve emptying of the SFP by mid-2032. The 7 years again is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*

3. ***Pre-Shutdown 5-Year Offload.*** *This offload scenario also includes two offload campaigns to dry storage prior to Unit 2 shutdown. The pre-shutdown campaigns are to occur after the last two refueling periods, the first of which occurs in 2021. Beginning in early 2030, almost 5 years after Unit 2 shutdown, a third and final offload campaign is performed to achieve emptying of the SFP in the fall of 2030. The 5 years again is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*
4. ***Pre-Shutdown Vendor Option Offload.*** *This offload scenario moves many more fuel assemblies than offload scenarios 2 or 3 prior to the Unit 2 end-of-life (EOL) shutdown in late August of 2025. It is the most effective offload scenario in expediting the reduction of radiological material in the SFP during reactor operation. However, it takes longer (until the end of 2033) to fully empty the SFP.*

Figure S-1. displays the scenarios graphically by illustrating the time-dependent number of fuel assemblies in the Unit 2 SFP for each offload scenario.

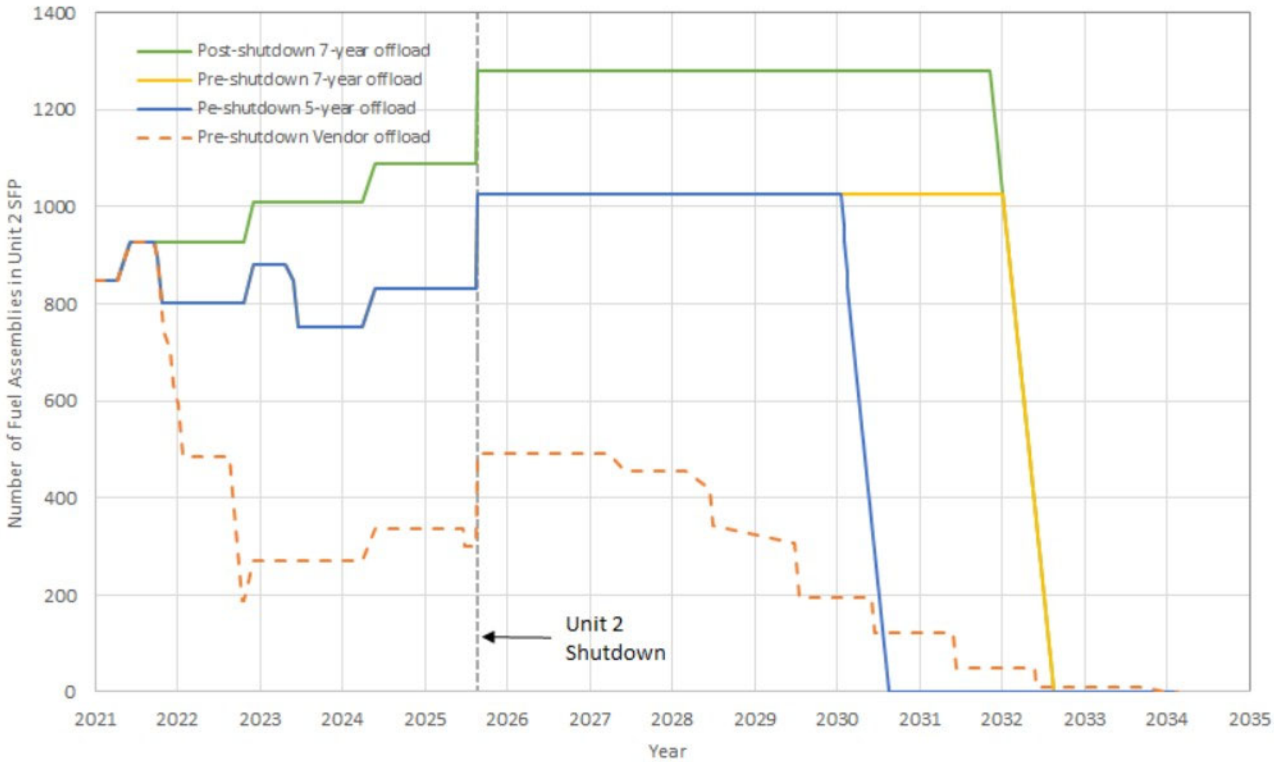


Figure S-1. Time-Dependent Number of Fuel Assemblies in the Unit 2 SFP Versus Calendar Year for Four Offload Scenarios

The fuel handling process assessed consists of the SFP, the ISFSI, and all the activities, structures, and equipment involved in storing and moving the spent fuel from the SFP and emplacing it in the ISFSI. Besides the SFP and ISFSI, structures include the fuel handling building (FHB) and auxiliary building (AB). Equipment items are the multipurpose container (MPC), the transfer cask, low-profile transporter (LPT), vertical cask transporter (VCT), cask transfer facility (CTF), and the storage overpacks. The displays of refueling outage times and number and timing of campaigns transferring the fuel assemblies to the ISFSI in Figure S-1 are tentative. Only one such offload scenario is to be selected for implementation and its specifics are likely to be revised to fit plant operational needs as the time for implementation draws nearer. The final offload scenario specifics are not expected to affect the results of this study.

The Risk Framework and Context

When questions of risk are involved, it is prudent to apply the principles of probabilistic risk assessment (PRA) to the extent necessary to identify and quantify the risks and the contributors. The result is increased knowledge for adopting best practices for managing the risks. The strategy of this analysis is to employ the principles of PRA to rank the four offload scenarios in terms of their safety risks. As noted, based on previous studies, all of the offload scenarios are considered to be safe. The question to be answered is which of the four provides the greatest amount of safety to the public and whether the differences in risk between them are significant. In particular, among the questions that will be answered in this risk assessment are the following.

1. *Which offload scenario is the least vulnerable to a beyond design basis radiological event, that is, which scenario of the four considered is most likely to assure public safety? How do the four offload scenarios rank?*
2. *How does the difference in risk between the SFP and the ISFSI impact the safest operating strategy?*
3. *What additional technology is needed to perform a full scope probabilistic risk assessment of the SFP and the ISFSI with the same confidence as those typical of contemporary nuclear power plant PRAs, particularly with regard to the quantification of the uncertainties?*
4. *How does the current spent fuel risk analysis compare the spent fuel risk with the DCPP risk of reactor core damage?*
5. *What are the most important variables affecting SFP risk?*
6. *How do the uncertainties in the assessment impact the conclusions?*

An overarching question is, "what do we mean by risk?" This study adopts the "triplet definition of risk" (Kaplan and Garrick, 1981; Garrick, 2008) as the framework for performing the analysis. This is the same framework used by the U.S. Nuclear Regulatory Commission (USNRC). That framework is founded on the premise that when one asks the question, what is the risk of something, they are really asking three questions: what can go wrong, how likely is it to go wrong, and what are the consequences. The "what can go wrong question" is answered by a set of potential accident sequences or scenarios; not to be confused with the offload scenarios described above. This has resulted in a theory of structuring scenarios that has evolved for years with great success in quantifying the risks of hundreds of complex systems involving hazardous materials. The question of "what are the consequences" is the aggregation of the end states of the "what can go wrong" scenarios. Finally, the question of "how likely is it to go wrong" is answered based on the supporting evidence and represented probabilistically to account for the uncertainties involved. The result is an "evidence based" risk assessment. Thus, risk assessment is the development of scenarios, likelihoods, and consequences that translate into not only what the risks are, but the contributors to the risk as well to enable corrective actions. The primary building block of a risk assessment is the set of "scenarios" of "what can go wrong." It is important to realize that a complete set includes scenarios that may end up having no human safety consequences, moderate consequences, or catastrophic consequences.

Understanding what we mean by risk is critical. Also critical is understanding why we analyze event scenarios beyond what the USNRC considers necessary to meet Federal safety requirements. This is sometimes referred to as "beyond design basis" (BDB) event (accident) analysis. This study is a good example of why this is done. There are four offload scenarios all of which it is believed can be demonstrated to comply with Federal safety requirements. But the desire is to know to what degree they are safe, and which one in fact is the safest based on assessments in terms of radiological consequences to the public. To obtain the desired level of resolution between the offload scenarios, it is necessary to turn up the microscope to include events not only within the design basis but hypothesize events beyond the standard set to detect significant differences in their risks. These are not events that are expected to occur but cannot be completely discounted as being impossible. An important lesson learned from the Fukushima accident is to consider BDB events (NASEM, 2016).

A fundamental requirement in any risk assessment is how the radionuclide source term is defined. Source term in this study is defined as the types and amounts of radionuclides that can potentially be released from an operation as a result of an undesired specific sequence of events, that is, an accident sequence. Thus, consequences have to somehow be connected to the fuel assemblies—the source of radionuclides. There is variation in radionuclide inventory among the fuel assemblies for which there also has to be accountability. The release of long-lived radionuclides of cesium [Cs-137 (30.2-year half-life) and Cs-134 (2.1-year half-life)] are used as surrogates for health effects. Previous studies of spent fuel accidents indicated cesium to be the dominant contributor to public risk. Transport of fuel assemblies from the SFP to the ISFSI does not occur until at least 5 years after a fuel assembly has been removed from the reactor, which provides time for the shorter-lived radionuclides to decay to levels having only a marginal impact on dose. In particular, the number of fuel assemblies overheated at any point of the offload process weighted by their cesium inventory constitutes the source term for the potential accident sequences in the assessment of spent nuclear fuel (SNF) risk.

Study Approach, Boundary Conditions, and Key Assumptions

The fact that each offload scenario involves the exact same operational steps in moving fuel from the SFP to the ISFSI greatly simplifies the analysis required for a risk ranking. That is, the goal of the study can be achieved by comparing offload risks rather than assessing individually the risk of each of the four scenarios. There are differences in the offload scenarios, but their impact in terms of risk are minimal and only lead to modest changes in the risks that determine their risk ranking. The differences mainly have to do with the time at which a particular offload campaign occurs, the number of fuel assemblies present, and the radionuclide inventory in the offloading step in which there is an event. In addition to the boundary condition of a comparative risk assessment, BDB basis events were purposely considered to enable making the differences in the safety of the offload scenarios more transparent. For example, rather than terminate the analysis at the threat of a 2.1 g spectral acceleration (SA) earthquake [the design basis of the DCPP (PG&E, 2018a)] consideration is given to much stronger earthquakes to achieve clear resolution of the risks of the four options for offloading the fuel.

Besides the two important boundary conditions of "comparative risk" and "beyond design basis" events, the approach was to focus on issues that can have a major impact on risk while contributing to the efficiency of the assessment. Key assumptions were made on screening initiating events, modeling of seismic impacts, the effects of reactor operation on the SFP, heavy load drops, surrogates for health effects, and the exclusion of events that are known not to be significant contributors to risk. Many other assumptions were involved in the assessment, but these are the dominant ones. Appendix B documents the complete list.

A review was made of a complete set of initiating events, both with respect to system events and failures due to natural and external events such as seismic, fires, aircraft impact, severe storms, and floods. While all known initiators were reviewed, it became clear that the offload risk was driven by two events; seismic events including those beyond the design basis (greater than 2.1 g SA) and heavy load drops onto the SFP. Three critical lifts are involved in all options for offloading spent fuel. The first two involve lifting the empty MPC inside the transfer cask (~96 tons) into the SFP and later lifting the fuel assembly loaded MPC inside the transfer cask (~117 tons) from the SFP for decontamination over the pool. The third critical lift is lowering of the loaded MPC (~45 tons) from the VCT into the storage overpack (~135 tons) which is mostly below grade at the CTF. Accidents during the third critical lift involve considerably less consequences as a result of an extreme event because of the much smaller radiological source term involved. Thus, it was not assessed further.

These two initiators (seismic events and heavy load drops) represented all the initiators evaluated in the assessment of SFP risks. It was also assumed that a concurrent severe reactor accident and a large early release resulting from a BDB seismic event may prevent onsite operator actions from mitigating the impacts of the same seismic event on the SFP or ISFSI. With respect to the heavy load drops at the SFP, it was assumed that a random drop over the SFP of a fully loaded transfer cask (~117 tons) would potentially lead to the failure of the SFP liner and concrete structure and a loss of SFP coolant. As to which initiating event dominates, it was found that for each of the four offload scenarios, the frequencies of dropped cask events are only a small fraction of the seismic initiated SFP fuel uncovery frequency. Both the seismic and dropped cask events that contribute to the risk have extremely low frequencies and are beyond the design basis of the DCPP. As noted earlier, BDB events were considered to make the risk differences between offload scenarios more transparent.

One major challenge of this risk assessment making it different than most is that the system being analyzed changes with time, unlike a stationary nuclear power plant. That is, the offload scenario involves different radionuclide source terms, depending on the time of the storage and handling activities in the process and the offload scenario being implemented. There are simplifications that can be made, many of which have already been noted. For example, the strategy of the analysis is to base it on all events, including extreme events that are BDB of the DCPP. What that means is to examine the whole offload scenario and pinpoint the activity having the greatest potential for a BDB event. That potential is locked up in the radionuclide source term. The result is the need to examine the source term as a function of time and activities in the total offload scenario. The other component of importance is the frequency of events at each activity which in combination with the source term (an indicator of the consequence potential) provide an indication of the risk as a function of activity.

Normally, the procedure would be to divide the total process into segments, analyze the risk of each segment of the process, and then integrate them over the total process. This would be done for each of the four offload scenarios, which would then be ranked in terms of their risks. Again, given that the goal is to compare the risks of the offload scenarios and that the process steps are the same for all the scenarios, more simplifications are possible. The simplifications are to put the emphasis on those activities that tend to bound the potential for BDB events. A cursory examination of the entire process made it clear that the activity dominating the risk was that having to do with the storage and removal of fuel assemblies out of the SFP. The risk of transport from the SFP and storage at the ISFSI is less. Also, transport and ISFSI risk do not differ much by offload scenario.

Another major challenge in performing a risk assessment of the DCPP offload scenarios was the development of a risk measure on which to compare the evaluated risks. The standard risk measures such as the frequency or probability of different levels of damage did not match well to the available information and the non-stationary nature of the system being analyzed. Thus, it was necessary to create a risk measure specialized to the offload process. The measure conceived is based on the limitations of the assessment and the characteristics of the offload process. The necessary building blocks on which to establish an offload scenario risk measure included the definition of a meaningful damage state, an appropriate radiological source term, accountability of the duration and time dependence of the radiological source term, and the driver of releases from the source term. The damage state adopted was the frequency of spent fuel uncovery within the SFP. The initial radiological source term was based on the decay of cesium since the fuel was offloaded from the reactor, and the driver for cesium release was overheating of the fuel. That is for a given accident, fuel uncovery, cesium inventory, time dependence, and fuel overheating become the staples for the risk measure. In particular, the risk measure is defined as "*the probability of fuel uncovery weighted by time (and duration) and the equivalent number of fuel assemblies which overheat and release cesium (adjusted for cesium decay), given a severe event including beyond design basis events.*"

The equivalent number of fuel assemblies are obtained after adjusting for the radioactive decay of cesium-134 and cesium-137 for the fuel assembly cooling times since offloaded from the reactor. In particular, the number of fuel assemblies at any point of the offload process that are vulnerable to overheating weighted by their cesium inventory constitutes the source term—again cesium being the surrogate for all radionuclides released. In the first few weeks after a reactor offload, shorter-lived radionuclides (e.g., iodine-131) in the fuel assemblies just offloaded can also potentially contribute to offsite public health effects. There is no offloading while the short-lived isotopes would contribute significantly to public health.

Key Findings and Risk Contributors

The risks associated with spent fuel pool operations bound the total risk of each offload scenario. This is primarily a result of the activities associated with storage and removal of the fuel from the spent fuel pool and the unique potential of the SFP source term—the total inventory of fuel assemblies present, their heat load, and the need for active cooling. The extent of radioactivity release given an SFP fuel uncovery event is a function of the total SFP heat load, not just the number and cesium inventory of fuel assemblies present. Of course, the ISFSI will contain more fuel assemblies than the SFP, many more in fact, but each fuel assembly stored at the ISFSI has less heat load, generally less specific radionuclide activity, and no requirement for active cooling. That is, ISFSI cooling is provided passively. Nevertheless, it is necessary to consider the consequences of losing passive cooling. A review of events, including BDB events, that could result in impeding convective cooling of the dry casks in the ISFSI did not result in a sufficient number of fuel assemblies being exposed as to compete with the seismic and heavy load drop events involving the spent fuel pool. The threat of a rockslide interfering with the passive cooling of the stored spent fuel casks has been evaluated with the conclusion that the hillslopes are stable. Although not impossible, a high acceleration seismic event during conditions of a water saturated hillside is a low frequency event. Seismic initiated landslides potentially burying a number of storages overpacks at the ISFSI are therefore included in the assessment of risks from SNF dry storage at the ISFSI.

Seismic events are determined to be important contributors for DCPP reactor-power operation and for the evaluation of DCPP SFP fuel uncovery frequency. The different fuel offload scenarios are not expected to have different seismic initiated frequencies that lead to SFP fuel uncovery.

Table S-1 presents the frequencies of significant accidents for three locations (SFP, handling and transport operations, ISFSI). The highest risk of a significant release is when the spent nuclear fuel is within the FHB at the SFP. Nearly all of the accident frequency derives from BDB seismic events. The transport of fuel assemblies from the FHB to the ISFSI has a much lower frequency of a severe accident; only about 1% of the frequency of events involving the SFP. The frequency is for the transfer of all remaining fuel assemblies from both the Unit 1 and Unit 2 SFPs. This low frequency result during transport comes from the design of the transport casks and the administrative controls in place at DCPP. Also impacting the low risk is the short exposure time of the operation. Transport operations represent only a small fraction of a year and at most only one MPC loaded with fuel assemblies for transport.

The frequency of significant accidents while in storage at the ISFSI is estimated to be greater than the handling and transport operations, but less than the SFP. The ISFSI risk considers severe earthquakes and includes aircraft impact and hill slides. None of the extreme threats is judged to result in damage to more than a few MPCs.

Table S-1. Contributors to Risk at Three SNF Locations

Contributor Groups	Frequency of Fuel Uncovery while SNF is in Unit 2 SFP (per year)	Frequency of Fuel Assembly Mechanical Damage During Transport from FHB to ISFSI	Frequency of Fuel Loss of Cooling while in Storage at ISFSI (per year)
Non-Seismic Initiators	3.2E-9 per year	2.2E-7	1.026E-6
Seismic Initiators	1.74E-5 per year	Screened as Negligible	4.80E-7 per year
Total Frequency at Each Location for All Initiators	1.74E-5 per year	2.2E-7 per year	1.51E-06 per year

Of the four offload scenarios, the vendor option offload scenario is very different from the other three. This is because its largest offload campaigns take place before the last reactor shutdown, or EOL of the plants, whereas the largest campaigns for the other three offload scenarios occur in the final campaign after EOL with no fuel assemblies left in the Unit 2 SFP. The pre-shutdown vendor offload scenario has a total of nine separate campaigns, three times any of the other three offload scenarios. The last four of the nine campaigns involve one or at most two MPCs to be transferred to dry storage. While the pre-shutdown vendor offload scenario moves a greater number of low decay heat fuel assemblies out of the Unit 2 SFP at the earliest possible dates, the total duration with fuel assemblies still in the SFP is longest. The reason is that the final fuel assemblies to be transferred must be further cooled to comply with the MPC design limits of total heat load. In particular, the vendor option offload scenario is 13 years, as measured from May 2, 2021, versus 9.3 years for the pre-shutdown 5-year offload, and 11.3 years for the other two offload scenarios. The result is an increase in the time that fuel assemblies are still in the SFP. This increases the chance that a large seismic event occurs with fuel assemblies present in the SFP for the pre-shutdown vendor offload scenario.

Because of the different timings of offload campaigns to dry storage, the peak number of fuel assemblies in the Unit 2 SFP varies among the four offload scenarios. The peak number of fuel assemblies (1281) is for the post-shutdown 7-year offload since it offloads no additional spent fuel assemblies until the Unit 2 EOL. The smallest peak number (928) is for the pre-shutdown vendor offload scenario.

Acceptability of Risk

The framework for the acceptability of the results of the risk assessment is rooted in the USNRC Safety Goals (USNRC, 1983) and in particular the quantitative health objectives (QHO). The QHOs are used as a basis for putting in context the computed risks from nuclear power plant operation. Compliance with the QHOs is viewed as a basis for establishing that the risks posed by the SFP operation are safe in comparison to the QHO numerical limits and therefore acceptable. The numerical QHOs invoked are not viewed as hard limits. The risks from the operation of the SFPs at DCPP for all four offload scenarios are concluded to be in compliance with the QHOs defined by the USNRC.

Safeguards Against Beyond Design Basis Events

The key safeguard to an initiating event that results in an initial loss of SFP cooling or inventory reduction is the availability of makeup water which when aligned also provides fuel cooling. At DCPP, the refueling water storage tank (RWST), condensate storage tank (CST), fire water storage tank (FWST), and raw water reservoirs (RWR) are seismically qualified, essential long-term cooling water supplies. Added to the multiple sources of water are (1) various options for aligning different sources of water to the SFP, (2) the ability to add fire protection water to the SFP when normal makeup water is not available, (3) the ability to use a fire engine when normal makeup systems fail to perform their intended functions, (4) the ability to spray firewater over the SFP for cooling when coolant level cannot be maintained in the SFP or if the area around the SFP is not accessible, (5) methods to manage the fire water system to ensure sufficient capacity is maintained to carry out vital functions, and finally, (6) the use of FLEX[2] equipment to assure such other needs as providing AC power for the equipment needed to perform the necessary extreme actions to assure SFP cooling.

Discussion of the Results

The results are provided as responses to the questions posed earlier for this assessment. The questions and responses follow.

1. *Which offload scenario is the least vulnerable to a beyond design basis radiological event, that is, which scenario of the four considered is most likely to assure public safety. How do the four offload scenarios rank?*

The ranking of the four offload scenarios is based on a risk measure defined explicitly for the handling and storage of *DCPP* spent nuclear fuel within the SFP. In particular, the measure is defined as "*the probability of fuel uncovery weighted by time (and duration) and the equivalent number of fuel assemblies which overheat and release cesium (adjusted for cesium decay), given a severe event including beyond design basis events."* This risk measure accounts for the following risk characteristics.

- The frequency of spent fuel uncovery within the SFP
- The duration and time dependent number of fuel assemblies in the SFP
- The extent of overheating of the fuel assemblies, given fuel uncovery
- The decay of cesium since offloaded from the reactor

The risk ranking of the offload scenarios is as follows. The numbers in the parentheses are the risk measure results for each offload scenario.

- Pre-Shutdown Vendor Offload (0.036)
- Pre-Shutdown 5-Year Offload (0.056)
- Pre-Shutdown 7-Year Offload (0.065)
- Post Shutdown 7-Year Offload (0.067)

[2] "FLEX" is an approach developed by the nucler industry to have flexible and diverse strategies for increased defense-in-depth for sequences involving an extended loss of AC power with a loss of normal access to the ultimate heat sink.

The pre-shutdown vendor option offload has the lowest risk measure. Its risk measure is, however, less than 50% lower than the risk measures of the other three offload scenarios. The risk measures do not differ substantially, especially when considering the uncertainties discussed later.

The above risk measure results were obtained by a time dependent integration over the period while fuel assemblies are present in the SFP. Figure R-1 illustrates this time dependent integration for each of the four offload scenarios. The integration begins with the start of the refueling scheduled for May 2021 and ends when the last fuel assembly is removed from the SFP. The date when the SFP is completely emptied of fuel assemblies varies with the offload scenario.

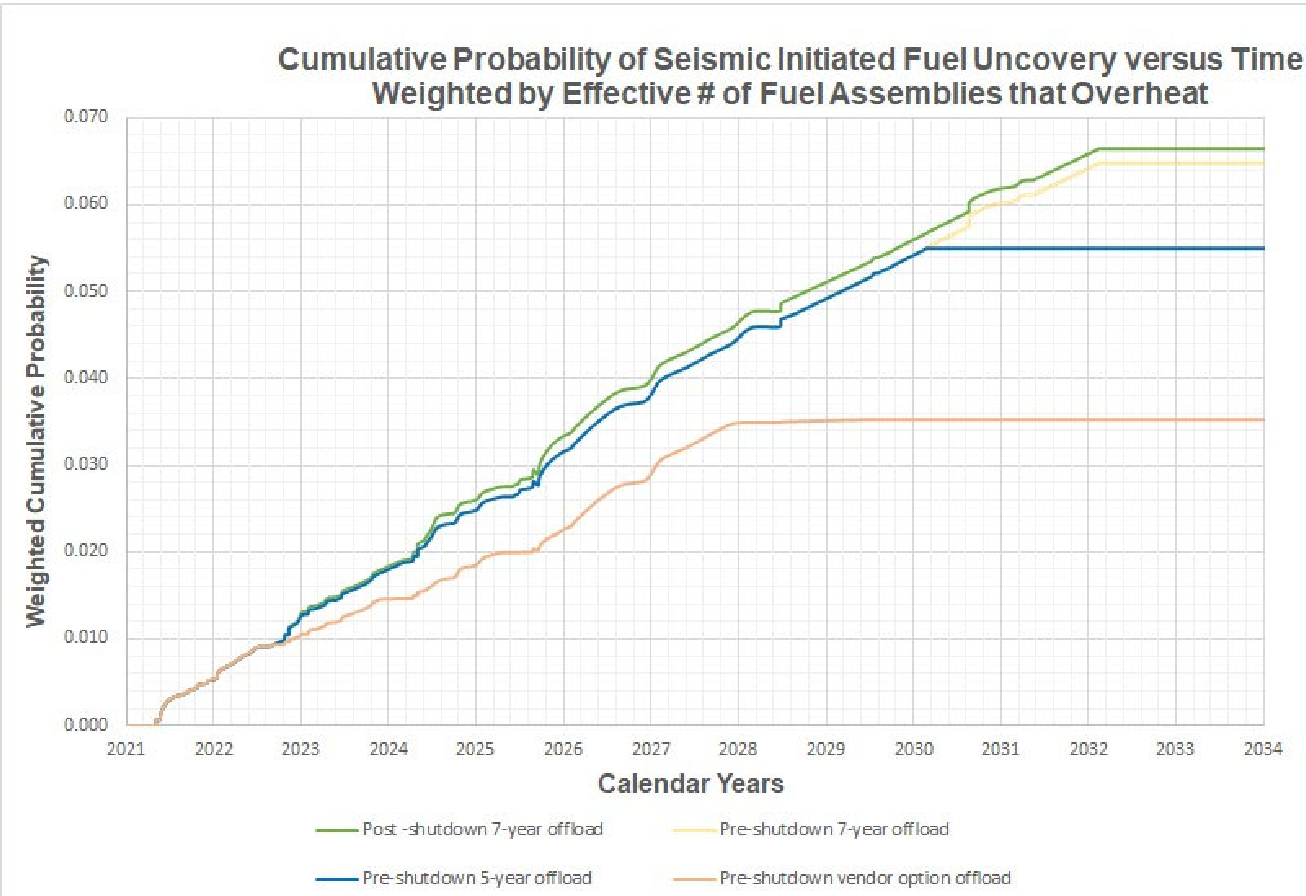


Figure R-1. Cumulative Probability of Seismic Initiated Fuel Uncovery Versus Time Weighted by the Number of Fuel Assemblies Overheated in SFP and Adjusted for Cesium Decay

Figure R-1 compares the weighted cumulative probability results for the four offload scenarios. The weighting for each interval in the integration is by the equivalent number of fuel assemblies that overheat at the time of the accident and adjusted for cesium decay since reactor offload. The weighted cumulative probabilities at year 2034 matches the base case risk measure results noted above. This shows that the pre-shutdown vendor option risk separates from the other three offload scenarios before the Unit 2 EOL full core shutdown (scheduled for August 2025) and remains below the other offload scenarios throughout. The pre-shutdown vendor option stops increasing after year 2028 when the SFP total heat load is insignificant, even though a small number of fuel assemblies remain in the SFP. The credit taken for passive heat removal by air cooling of the uncovered fuel assemblies within the SFP means that no overheating occurs once the total and individual fuel assembly heat loads drop below specified levels.

Figure R-2 presents for each of the offload scenarios the probability of an SFP fuel uncovery event with a given amount of cesium release or greater.

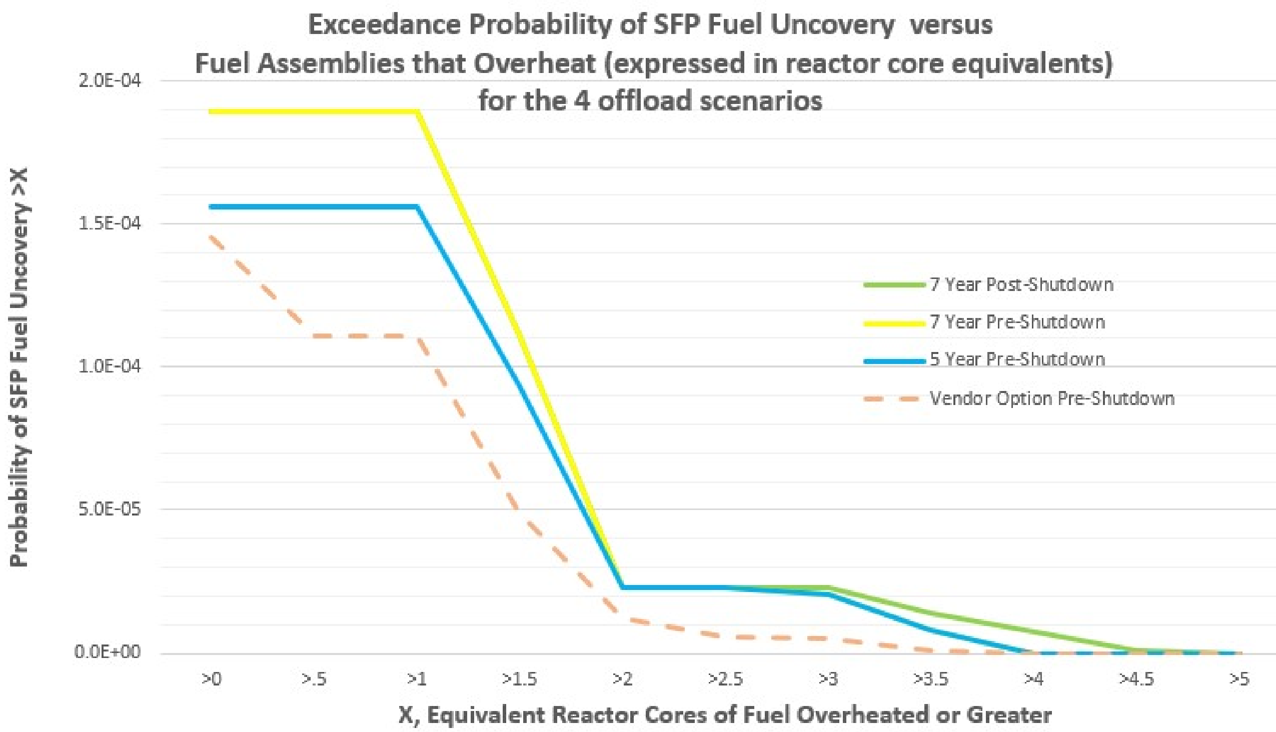


Figure R-2. Probability of Seismic Initiated SFP Fuel Uncovery while Fuel Assemblies are in the SFP Versus Number of Equivalent Reactor Cores of Fuel Assemblies that Overheat

Accountability of the radiological source term provides a different perspective of the risk. For example, the public health consequences of a release from several fuel assemblies would be much less than an event involving a large number of fuel assemblies. It is important to note the exceedance feature of this graphic. By this technique, the integrated fuel uncovery probability for releasing more cesium than the equivalent of 1.5 reactor cores for the vendor option offload scenario is just 5E-5[3]. The fuel uncovery probability for releasing more cesium than the equivalent of more than one reactor core, again for the vendor option, is about 1.1E-4. The maximum value on the horizontal axis is less than the peak number of fuel assemblies in the SFP for all offload scenarios because not all fuel assemblies present are vulnerable to overheating and because cesium decay has been taken into account.

Figure R-2 indicates that the vendor offload scenario exceedance probability is always less than the other offload scenarios.

2. *How does the difference in risk between the SFP and the ISFSI impact the safest operating strategy?*

The SNF is less of a threat to the public on a frequency basis when in dry storage at the ISFSI compared to wet storage in the SFP; only 7.5% (Table R-1) as much risk when comparing the frequencies for each offload scenario of ISFSI potential fuel overheating with the frequencies of SFP

[3] The convention adopted is to interpret NEn as N x 10^n, n may be plus or minus. Example 5E-5 is 5 x 10^{-5}.

fuel uncovery. The risks to the public by transfer of SNF from the FHB to the ISFSI is small compared to the risk at both storage locations. The DCPP total frequency of SFP fuel uncovery of 1.74E-5 per year is noticeably higher by factors of 3 to 10 than the SFP fuel uncovery frequencies developed for other sites (see Table 4-2). The higher seismic initiator contribution for DCPP is a result of the much higher seismic hazard. The more seismically robust design or the SFP liner and structures limits the seismic fuel uncovery frequency increase.

A more direct comparison of the risks of the two storage locations follows. This comparison is between the risk reduction from the SFP and the risk increase at the ISFSI when transferring a number of MPCs loaded with fuel assemblies two years earlier than planned (See Section 8.4). It is found that even a bounding assessment of the increase in risk measures at the ISFSI is less than 1% of the decrease in risk measure for the SFP. The offload scenario risk measure rankings show that transferring fuel assemblies earlier to the ISFSI decreases the risk at the SFP, and this decrease is only minimally offset by the increase in the risk measure at the ISFSI. As noted in the response to Question 1, the differences in risk between the four offload scenarios considered are modest. There are also heat load limits on the MPCs used to transfer SNF to dry storage which limit the amounts and earliest times that fuel assemblies can be transferred to the ISFSI.

Table R-1. Comparison of Initiating Event Frequency Contributors (events per year) Between Fuel Unrecovery at the SFP and Potential Fuel Overheating while SNF is at the ISFSI Pads

Seismic Frequency per year of Fuel Uncovery while SNF is in Unit 2 SFP (per year)	Random Frequency per year of Dropped HI-TRAC onto SFP causing Fuel Uncovery, 2 lifts per MPC Transferred	Total Frequency of Fuel Uncovery per year at SFP	Frequency of One or More Storage Overpacks With Potential for Fuel Overheating at ISFSI Pad	Ratio of Storage Overpacks Potential Overheating Frequency to SFP Fuel Uncovery Frequency (%)
1.74E-05	3.20E-09	1.74E-05	1.31E-06	7.5%

3. *What additional technology is needed to perform a full scope probabilistic risk assessment of the SFP and the ISFSI with the same confidence as those typical of contemporary nuclear plant PRAs, particularly with regard to the quantification of the uncertainties?*

In this assessment of risks at the SFP and ISFSI, the following technical limitations were encountered.

For the SFP:

a) A realistic development of the seismic capacities is required of the SFP structure and liner, including considerations of their likely failure modes and detailed knowledge of the liner openings and potential drainage paths through the concrete structure. At present the high confidence of low probability of failure (HCLPF) acceleration for the seismic capacity curve has been evaluated but not the full capacity curve.

b) An approach needs to be developed for evaluating the potential for air-cooling by natural convection within the SFP and FHB to determine the extent of fuel overheating and for assessing the resulting radiological releases from uncovered fuel to the environment, i.e., an SFP- specific MELCOR (Sandia, 2017, Sandia, 2017a) type modeling capability is needed.

c) An assessment of the potential for fuel overheating, given fuel uncovery and the resulting radiological releases is required.
d) The absence of severe accident sequence-specific calculations of the onsite radiation environments precluded knowing the effects they would have on the ability to perform the directed mitigating operator actions. A modeling assumption was made to limit such credit for sequences judged to involve high radiation releases from the reactor core. Recovery actions to respond to events involving only the SFP should occur before any radiation releases from the fuel assemblies within the SFP.
e) A detailed assessment is needed of the potential impacts of seismically initiated severe accidents involving the reactor during power operation. The Fukushima accident revealed the potential for hydrogen releases from the reactor containment into nearby plant locations containing SFP cooling equipment.

For the ISFSI:

a) A realistic evaluation of storage overpack and concrete pad seismic capacities is needed including accelerations well beyond the design basis. An assessment of seismic capacity to the point of failure or to very low seismic frequency is desired. Failure modes which could lead to the overheating of stored fuel assemblies are of interest.

b) Extension of the thermal analysis performed to date involving the storage overpack is required to account for the decreases in radiological inventories and decay heat levels for different times after reactor offload. The licensing calculations consider only the most bounding heat loads.

Technical limitations for both the SFP and ISFSI:

a) Many more load lifts have been conducted since the original heavy load drop frequency per movement was assessed. This additional evidence could be included in a reassessment of the heavy load drop frequency.

b) DCPP site specific modeling of time-dependent radiological releases from either storage location, transport by winds offsite, and the effects on the public considering evacuation would be useful. At present bounding release assumptions used in the USNRC's generic site modeling are used for comparison against the USNRC's QHOs.

4. *How does the current analysis compare the spent fuel risk with the DCPP risk of reactor core damage?*

As noted in Table R-2, the mean frequency of events initiating fuel uncovery in the SFP is about 22% of the total reactor core damage frequency.

Table R-2. Severe Accident Frequencies and Recurrence Intervals for DCPP Unit 2

Severe Accident Frequencies and Recurrence Intervals for DCPP Unit 2	**Frequency (events per year)**	**Recurrence Interval (years)**
Unit 2 Seismic SFP Fuel Uncovery	1.74E-05	57,000
Unit 2 Total Reactor CDF	8.15E-05	12,000
Unit 2 Total Reactor LERF	7.90E-06	127,000
Seismic Only Unit 2 Reactor CDF	2.83E-05	35,000
Seismic Only Unit 2 Reactor LERF	5.17E-06	193,000

The seismic initiated mean recurrence interval for SFP fuel uncovery (about 57,000 years) is about five times the DCPP Unit 2 total core damage frequency (CDF) mean recurrence interval of 12,000 years; i.e., the frequency is lower for the SFP fuel uncovery than for the reactor CDF. The frequency is higher and so the recurrence interval for SFP fuel uncovery is less than half that for the Unit 2 reactor large early release events (127,000 years). In contrast to this assessment of SFP fuel uncovery risks, only about one-third of the reactor CDF comes from seismic events and two-thirds of the reactor large early release frequency (LERF). The comparison with the severe reactor accident LERF in Table R-2 is for information only.

The comparison of SFP risk and nuclear plant risk requires an assessment of whether the SFP fuel uncovery releases would be large. Whether all fuel assemblies contribute to the radiological release during an SFP fuel uncovery depends on the effectiveness of air-cooling by natural convection and potential SFP spray effectiveness in removing fission products from the effluent stream. This assessment has attempted to account for air-cooling by natural convection within the SFP in assessing the extent of radiological release given uncovery of the fuel assemblies. As noted earlier, this is an area requiring further study.

5. *What are the most important variables affecting SFP risk?*

Analyses indicate that the most important variables to SFP risk are the human error probabilities (HEP) assigned for high acceleration seismic events. The operator actions accounting for the HEPs have to do with the proper alignment of alternative paths for SF cooling and for SFP coolant makeup. These actions are credited when the structures and equipment related to the SFP for these functions remain operational.

A key uncertainty for such sequences is the capability for air-cooling by natural circulation within the SFP that could limit fuel overheating, potentially precluding a radiological release. The conditions when air-cooling by natural convection would be successful without operator actions can greatly influence the total SFP risk for each of the four evaluated offload scenarios.

6. How do the uncertainties in the assessment impact the results?

There are several sources of uncertainty in the assessment. The uncertainties introduced by the assumptions listed in Appendix B are examples. Five important sources of uncertainty are shown below.

- Seismic hazard frequencies
- Seismic capacity of key structures
- Human error probabilities
- Heavy load drop frequency
- Extent of fuel overheating, given an SFP uncovery event

The spent fuel uncovery frequency is driven by seismic events, whose uncertainty is largely due to the uncertainties in the seismic hazard curves. The uncertainties in the seismic hazard curve are large and vary with the acceleration range. Typically, the uncertainties are largest for the extremely high accelerations. The spent fuel uncovery frequency is affected by the seismic hazard. In this regard, the frequency of spent fuel uncovery per calendar year with fuel assemblies in the SFP does not differ for the four offload scenarios and therefore does not affect their rank-order. There has been considerable effort in characterizing the seismic hazard and further analysis is not expected to change the ranking of the offload scenarios. The differences between offload scenarios in the remaining durations of spent fuel being stored in the SFP are modest. The pre-shutdown vendor option offload scenario has the longest remaining duration of spent fuel being stored in the SFP, but its final period of spent fuel storage is with a much lower number of fuel assemblies than the other offload scenarios. This is a reason why the risk metric chosen for comparison also considers the amount of radiation that could potentially be released by the extent of fuel overheating and not just by the frequency of spent fuel uncovery.

The capacity of the key structures such as the fuel handling and auxiliary building can determine the impact of seismic events on the level of severity of the accident—severity levels having reference to whether it involves a loss of cooling, a leak, or structural collapse. Structural collapses are likely to preclude access for recovery actions. These types of impacts affect the frequency of extended spent fuel pool uncovery. The impact of the cause of the fuel uncovery on radiological release is a secondary effect, thus a secondary effect on the ranking of the offload scenarios.

Uncertainties in HEPs dictate the recovery from the seismic impacts and therefore the spent fuel pool uncovery frequency. Credit for local operator actions is difficult to justify for large seismic accelerations. Again, while the uncertainties may be high, it is believed to have only a secondary effect on offload scenario ranking.

The uncertainties in heavy load drop frequencies are largely judgmental and are not fully based on event records. The range of uncertainty is estimated as a factor of ten lower and a factor of ten higher than that assumed. The contribution from drops is less than one in a thousand contribution to the SFP uncovery frequency. While there are statistical data on cranes that are not single failure approved, the data is yet to be processed for single failure approved cranes. This assessment indicates that heavy load drops leading to spent fuel uncovery are only a very small fraction of the total spent fuel pool uncovery frequency. Drops occurring at the cask transfer facility that may potentially result in mechanical damage to fuel assemblies have been analyzed to determine they would not result in fuel overheating. Frequencies of drops at the CTF are also very low. Thus, this uncertainty is not expected to affect the ranking of offload scenarios.

The uncertainties associated with overheating of fuel, given a fuel uncovery event, are significant and do impact the ranking of the offload scenarios. The extent of fuel overheating, given fuel uncovery, can be characterized by two attributes. First is the number of prior reactor offloads of fuel vulnerable to overheating and second, the capability of natural processes for heat removal in the absence of being able to provide cooling. Limited overheating of the fuel would minimize the differences in the offload scenarios. On the other hand, substantial overheating of the fuel would increase the differences in the offload scenario rankings. The changes in the risk measures would add to the preference of the vendor option from a risk perspective. However, changing the assumption about the extent of fuel overheating would not change the fact that the risks from each of the four offload scenarios would still be well below the USNRC's QHO criteria.

What Does It All Mean?

The risks from the operation of the SFPs at DCPP for all four offload scenarios are concluded to be in compliance by substantial margins with the QHOs defined by the USNRC. The study therefore focused on the assessment of SFP risks that differ between the four offload scenarios. The study also confirmed that the risks associated with all of the spent fuel offload scenarios are extremely small, approximately half to an order of magnitude below most of the accident scenarios associated with Units 1 and 2 core damage frequencies. The contributing events hypothesized are all well beyond the design basis of the DCPP and its current licensing requirements. Essentially all of the activities for transferring and transporting the spent fuel from the spent fuel pool to the dry cask storage facility are common to all four offload scenarios. The primary difference is the schedule for the offload campaigns which impact the inventories of spent fuel and therefore the radionuclides potentially available for release at different times in the event of a severe accident. Accounting for the uncertainties simply associated with human performance alone could substantially lower the absolute frequencies but not alter the relative rankings of the offload scenarios. It is possible that events screened based on DCPP-specific conditions rather than the generic data used in this analysis could alter the conclusions. For example, how sensitive to human performance is performing the offload in one larger campaign versus nine different campaigns spread over several years, some of which are to take place while the nuclear power units are still operating? Quantifying this sensitivity would require more extensive study.

In spite of considering all of the standard internal and external initiating events generally considered for nuclear power plants and the dropped cask initiating event unique to spent fuel handling, only BDB seismic events resulted in any significant resolution of differences between the offload scenarios. While the offload scenarios have been ranked on the basis of safety to the public, their assessed risks are a factor of two within each other. The primary uncertainties are human performance under extremely severe accident conditions, threshold conditions for self-sustaining oxidation reactions of the fuel material, accountability of specific spent fuel assembly heat loads, and the fragilities of the fuel handling and auxiliary buildings.

1 Purpose and Background

The absence of an offsite Federal high-level waste disposal system and the practice of storing spent nuclear fuel (SNF) at reactor sites, together with the Fukushima accident resulting from the Great East Japan Earthquake on March 11, 2011, have brought spent fuel pools (SFP) and dry cask storage, and the associated activities into focus as requiring more rigorous safety assessment. Onsite storage has especially impacted the role of SFPs in terms of their inventories and operations. It has added activities to the nuclear power plant sites not originally intended, namely the higher density configuration of fuel assemblies in the SFP to accommodate greater numbers of fuel assemblies, more handling operations of the fuel assemblies, and several operations associated with moving spent fuel from the SFPs to the independent spent fuel storage installation (ISFSI), not to mention the safety of the ISFSI itself.

The purpose of this study to devlop a methodology for Probabilistic Risk Assessment of nuclear power plant spent fuel handling and storage programs and demonstrate its application and by assessing the radiological risks associated with storage and movement spent fuel from the spent fuel pools (SFP) to the Independent Spent Fuel Storage Installation (ISFSI) at the Diablo Canyon Power Plant (DCPP). This is done by evaluating the comparative risk of different spent fuel offload scenarios provided by PG&E. Offload scenarios refer to the strategies for relocation of the spent fuel from the SFP to the ISFSI. In particular, the assessment is guided by seeking responses to the following six questions:

1. *Which offload scenario is the least vulnerable to a beyond design basis (BDB) radiological event, that is, which scenario of the four considered is most likely to assure public safety? How do the four offload scenarios rank?*
2. *How does the difference in risk between the SFP and the ISFSI impact the safest operating strategy?*
3. *What additional technology is needed to perform a full scope probabilistic risk assessment (PRA) of the SFP and the ISFSI with the same confidence as those typical of contemporary nuclear power plant PRAs, particularly with regard to the quantification of the uncertainties?*
4. *How does the current analysis compare the spent fuel risk with the Diablo Canyon Power Plant (DCPP) at-power risk of core damage?*
5. *What are the most important variables affecting SFP risk?*
6. *How do the uncertainties in the assessment impact the results?*

Beginning in January 2021 and lasting until all fuel assemblies are offloaded from the DCPP spent fuel pool, the four alternative SFP offload scenarios are shown in Figure 1-1. These offload scenarios were developed for Unit 2 (HOLTEC International, 2012;TN Americas, 2019) and finally defined by PG&E. The offload scenarios are expected to be similar for Unit 1. All four offload scenarios assume Unit 2 final shutdown in late August of 2025. The displays of refueling outage times and number and timing of campaigns transferring the fuel assemblies to the ISFSI in Figure 1-1 are tentative. Only one such offload scenario is to be selected for implementation and its specifics are likely to be revised to fit plant operational needs as the time for implementation draws nearer. The final offload scenario specifics are not expected to affect the conclusions of this study.

The offload scenarios are described below.

1. ***Post-Shutdown 7-Year Offload.*** *No offloads to dry storage are performed while the Unit 2 reactor is operating. Beginning in late 2031, almost 6 years after Unit 2 shutdown, a single offload campaign is performed to empty the spent fuel pool by mid-2032. The 7 years is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*
2. ***Pre-Shutdown 7-Year Offload.*** *This offload scenario includes two offload campaigns to dry storage conducted prior to Unit 2 shutdown. The pre-shutdown campaigns are to occur after each of the last two partial core refueling periods, the first of which occurs in 2021. Beginning in early 2032, almost 7 years after Unit 2 shutdown, a third and final offload campaign is then performed to achieve emptying of the SFP by mid-2032. The 7 years again is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*

3. ***Pre-Shutdown 5-Year Offload.*** *This offload scenario also includes two offload campaigns to dry storage prior to Unit 2 shutdown. The pre-shutdown campaigns are to occur after the last two refueling periods, the first of which occurs in 2021. Beginning in early 2030, almost 5 years after Unit 2 shutdown, a third and final offload campaign is performed to achieve emptying of the SFP in the fall of 2030. The 5 years again is the time from Unit 2 shutdown to when the SFP is completely empty of fuel assemblies.*

4. ***Pre-Shutdown Vendor Option Offload.*** *This offload scenario moves many more fuel assemblies than offload scenarios 2 or 3 prior to the Unit 2 end-of-life (EOL) shutdown in late August of 2025. It is the most effective offload scenario in expediting the reduction of radiological material in the SFP during reactor operation. However, it takes longer (until the end of 2033) to fully empty the SFP.*

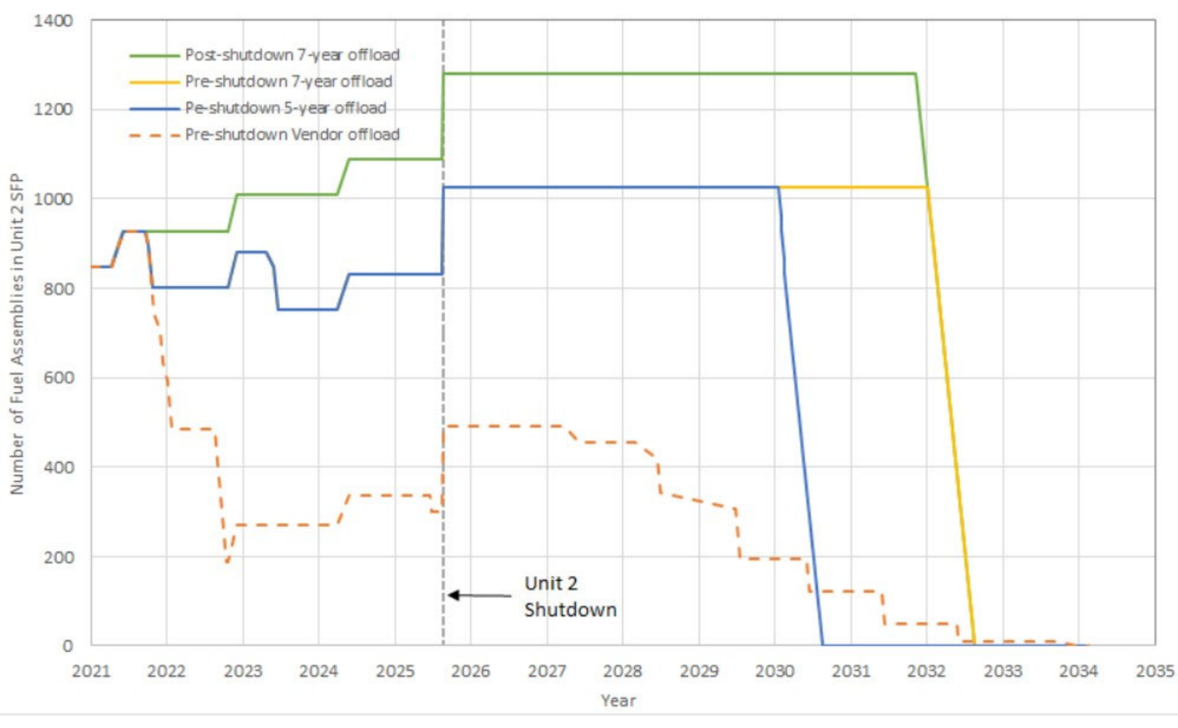


Figure 1-1 Diablo Canyon Power Plant (DCPP) Spent Fuel Offload Scenarios

While PRA principles are used in this study, it is not intended that this be a full scope quantitative risk assessment. On the contrary, the goal is a first approximation of the differences in risk between the four proposed offload scenarios described above. Simplifications of the assessment are taken to expedite obtaining the necessary risk insights to resolve the differences in the safety of the four scenarios and to respond to the questions concerning the risks of the SFP and the ISFSI. The key feature of this assessment that enables providing reasonable responses to the six questions is that this is a comparative study of the risks associated with the four scenarios, not a risk assessment of each individual offload scenario. This is made possible by the fact that each of the four scenarios involves the same basic activities, only the schedule for their implementation changes.

Another feature of the study enabling some quick insights on the risks involved is the use of event data readily available in the literature. More details on the simplification process and its validity are discussed in Section 3. Section 2 that immediately follows provides a high-level overview of the DCPP spent fuel handling process.

2 Overview of DCPP Spent Fuel Handling

The major structures, systems, and components of the DCPP ISFSI are the storage pads, cask transfer facility (CTF), onsite transporter, and dry cask storage system. The transfer of spent nuclear fuel from the SFP to the ISFSI involves, 1) several operations associated with the SFP, 2) transferring and transporting of spent fuel to the onsite ISFSI pad, 3) emplacement of the SNF into the ISFSI while performing the needed routine inspection and maintenance, and 4) eventually the operations associated with removing the spent nuclear fuel from the site. Step 4 is not a part of this study.

Figure 2-1 is a PG&E poster diagram of the offload operations. Briefly, the operations within the fuel handling building (FHB) containing the SFP include moving the transfer cask containing the stainless steel MPC into the SFP, transferring SNF assemblies into the MPC (multiple fuel handling crane operations), placing the lid on the MPC, lifting (~117 ton lift) the loaded MPC inside the transfer cask from the SFP for decontamination over the pool, moving the transfer cask with the loaded MPC to the washdown area for welding the MPC lid, replacing the water (dehydration) from the MPC with helium gas and installing an outer lid on the transfer cask. A low-profile transporter (LPT) is then used to bring the transfer cask outside the FHB.

A vertical cask transporter (VCT) is used to move the transfer cask/MPC assembly from outside the fuel handling building/auxiliary building (AB) (approximately 115 feet in elevation) to the cask transfer facility (approximately 310 feet in elevation), which is adjacent to the ISFSI storage pads on a hill above the DCPP. The VCT enables the exchange of the MPC from the transfer cask to the storage overpack for final emplacement on the storage pad. This operation is facilitated by the storage overpack being largely below grade in the CTF.

The loaded storage overpacks are then lifted out of the CTF (180 tons, which is the heaviest of all lifts outside), moved onto the ISFSI concrete storage pad and bolted into place on the pad within a protected area separate from that of DCPP. Each storage pad is designed to accommodate up to 20 loaded storage overpacks in a 4-by-5 array. There are seven such pads. Each loaded storage overpack is approximately 11 feet in diameter, 20 feet high, and weighs about 180 tons. There is approximately 6 feet, surface-to-surface distance between the storage overpacks. The series of seven storage pads cover an area approximately 500 feet by 105 feet.

All four offload scenarios involve the same operations as described and depicted in Figure 2-1. Offloading experience may expose improvement opportunities in the process. Current experience includes seven loading campaigns and 1856 SNF assemblies stored at the ISFSI in 58 casks.

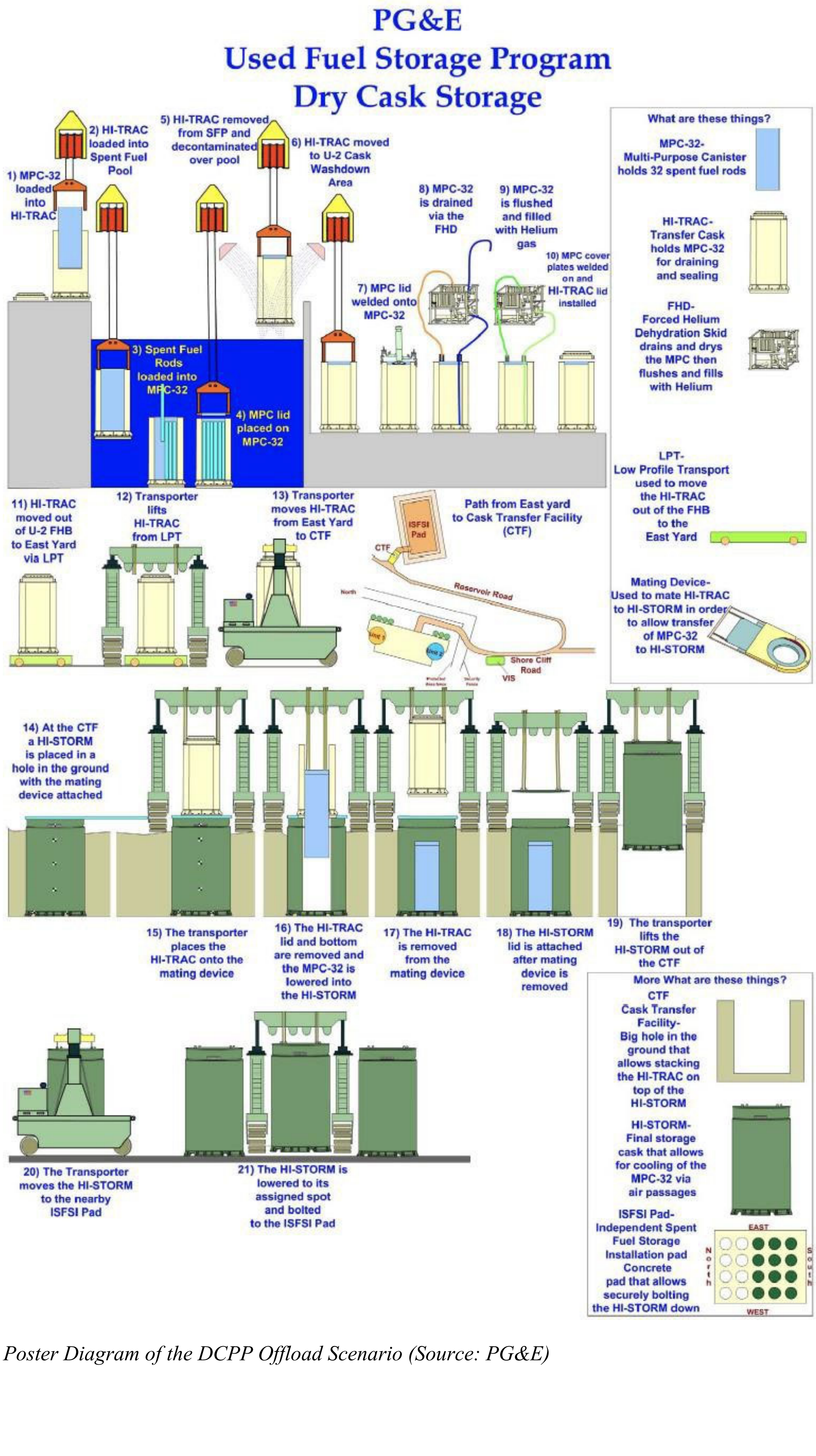


Figure 2-1. Poster Diagram of the DCPP Offload Scenario (Source: PG&E)

3 Risk Assessment Approach and Application

The principles of PRA (Garrick, 2008) are applied to assess the comparative risk of the four different DCPP spent fuel offload scenarios. The starting and endpoints of the four scenarios are the SFP and the ISFSI. Thus, questions of risk apply to the SFP, the handling and transfer operations for transporting and emplacing the spent fuel in the ISFSI, and the ISFSI itself. The result is a major and complex set of facilities and operations making up the system to be analyzed.

The good news is that there have been changes in the Federal regulations and increased application of PRA methods to assess spent pool and site-specific spent fuel dry storage risks (USNRC, 2001; USNRC, 2007; and EPRI, 2014). Assessing such risks should be feasible as the PRA methods are basic enough to apply to any kind of system. In fact, the "triplet definition of risk" methodology was established as the foundation for a general theory of quantitative risk assessment. The real challenge to assessing the risk of onsite spent fuel storage is understanding the underlying processes associated with the progression of accidents that may occur at the SFP, the ISFSI, and all the operations in between. More on this in Section 5.

Given that the goal is the comparative risk between the scenarios and between the SFP and ISFSI, as noted earlier major simplifications of the safety assessment are possible while preserving a PRA framework of analysis. The features supporting simplification are the following.

(1) All four scenarios involve essentially the same operations, procedures, equipment, and number of fuel assemblies; only the schedules for offloading change, i.e., the campaigns to transport the fuel assemblies to the ISFSI change. One minor difference in operations between offload scenarios is in the number of fuel assembly movements needed within the SFP to achieve the 1x4 fuel assembly configuration following reactor offload or following the loading of an MPC for transfer to the ISFSI. The 1x4 fuel assembly configuration (USNRC, 2013) is where cooler fuel assemblies are positioned around hotter fuel assemblies (Figure 5-1). The movement of single fuel assemblies is not risk significant. Thus, this difference between offload scenarios is neglected.

(2) The radiological inventories are well defined by location and operations; hence the radiological risk potential of the different operations is well defined.

(3) The primary goal is the assessment of risk of a major radiological release involving offsite consequences, thus enabling the screening of events not having a radiological inventory (source term) with the properties to potentially support a major release.

(4) Comparative risks enable the use of surrogates to radiological consequences, thus avoiding the need to perform complex radiological transport, dose, and health effects computations.

A review of the radionuclide source terms and their configuration over the entire spent fuel offload scenario of the DCPP suggests that the SFP is the only location where a large number of fuel assemblies exist requiring human intervention to mitigate a loss of SFP cooling (or coolant) and the prevention of fuel damage. Passive air cooling for SFP fuel is not adequate for all accident scenarios. Provisions for aligning a seismically qualified coolant makeup source to the SFP from a higher elevation have been made for just this reason.

In terms of severe consequences, the SFP risk of a severe accident may to a first approximation represent the risk of the entire offload scenario; i.e., the total risks of wet storage within the SFP, transport to dry storage, and the risks at the ISFSI. Obviously, the SFP risk is not the total risk, but has the potential to be the dominant risk, at least with respect to severe accidents that can potentially result in offsite consequences. However, reaching this conclusion even at a qualitative level requires some investigation of the initiating events representing a threat to any part of the total system, namely the SFP, the transfer and transport of the MPCs to the ISFSI, emplacement of the MPCs into the ISFSI, and the ISFSI itself. A screening of initiating events that apply outside the SFP is discussed in Section 4.

Section 4 provides a detailed discussion of potential initiating events divided into six groups. The groups are (1) heavy load drops, (2) losses of inventory of the SFP, (3) losses of SFP cooling, (4) spatially dependent internal events, (5) external events and (6) criticality events. Of course, the ISFSI risk is relevant as well because of its increasing number of fuel assemblies with time, albeit passively cooled. Because of the simplification features noted, many of the initiating events can be screened as discussed in Section 4. Analyses will have to be performed to estimate the risk of the ISFSI to provide insights on just how well the SFP risk represents the risk of the entire offload scenario. It is not believed that the radiological inventories represented by the handling, transportation, and emplacement of spent fuel in the ISFSI are significant candidates for a major radiological release, at least in comparison to the SFP or ISFSI. To be sure, they represent risks that are more likely to occur, but the consequences are much more limited than the risks associated with the SFP and the ISFSI.

The point has been made that the SFP is believed to be the only location where a severe accident could occur. But a risk assessment is based not just on belief, but on the supporting evidence and analysis of the total system. To answer questions about the least risk offload scenario, it is necessary to consider the risk of all the activities associated with the spent fuel offload scenario, which includes the various spent fuel handling operations of transferring, transporting and emplacing the spent fuel in the ISFSI. The final phase is the risk of the ISFSI itself.

Normally, to perform a full scope and rigorous PRA of such a complex array of activities, especially considering multiple offload scenarios, represents a major undertaking involving many months if not years and millions of dollars' worth of effort. We have limited the scope on the belief that we are not compromising the results for the intended purpose. That purpose is to make the right decisions on the risk management of the DCPP spent fuel program.

This limited scope short term DCPP spent fuel risk (DSFR) assessment is made possible by the extensive amount of work that has already been performed. The challenge is to put the pieces together in a logical way as to represent the risk of the total spent fuel handling system. We are fortunate to have the benefit of a comprehensive safety analysis of the spent fuel handling activities in the FHB (PG&E, 2018) and the Diablo Canyon spent fuel storage installation update for the final safety analysis report (UFSAR) (PG&E, 2018a) to draw from for assessing the risk. This is one of the reasons for a more thorough analysis of the operations involving the SFP.

The results from the ISFSI UFSAR are reviewed from a risk perspective and specialized to the offload process. For example, consideration is given to the exposure times of specific activities in the offload scenario and accounted for in the frequencies of events occurring that might damage the spent fuel. Also, the number of fuel assemblies as a function of the operation activity is considered along with their heat load and fuel burnup, that is, the radiological inventory as a function of the phase of the offload process.

The full spectrum of initiating events is considered for all three phases of the offload scenarios; SFP, spent fuel handling, and the ISFSI. The initiating events include heavy load drops, transit accidents, equipment failures, extremely severe seismic events including slope sliding, loss of offsite power, human failures and other external events such as aircraft crashes and explosions, all as a function of the different activities involved.

The assessment approach is to first screen the full set of initiating events down to those determined to most likely be the key risk contributors, and in particular to limit the selection to those likely to reveal different risks for the four postulated offload scenarios. The selected set of initiating events is then quantitatively evaluated for accident sequence frequencies using insights and data from generic PRA studies of SFP risk that have been performed to date. Where readily available, failure rate and hazard frequency information specific to DCPP and the DCPP site is used. Where not available, generic failure rate data is used. Fortunately, there has been major risk analysis work performed for the DCPP. Existing information for DCPP is used to supplement available structure seismic capacities to approximate seismic capacities for similar, but not yet analyzed structures. A full scope PRA would further attempt to use DCPP specific data and seismic capacities for all key risk elements. This approach, though providing numerical results, is to be interpreted as approximate results since a formal uncertainty assessment has not been performed. Nevertheless, the approach provides key risk insights and the identification of important assumptions that drive the results. The approach exposes the DCPP specific operating practices, available equipment, and procedures directing the operator's responses to potential initiating events.

The DCPP spent fuel risk analysis takes advantage of the several studies of SFP risk performed in the U.S. Several references are noted in Section 4. Once evaluated, the acceptance of the calculated risks is also of interest. The USNRC has performed much work in this area (e.g., USNRC, 2013a). The conclusions from that work which builds upon many previous studies developed specifically for severe accident SFP risks are summarized later in the report. Meanwhile, it is important provide a framework for the acceptability of the results.

In 1983, the USNRC presented its safety goals for nuclear power plant operation (USNRC, 1983). Qualitative goals were defined to ensure both individual risk and societal risk. The USNRC then defined two quantitative health objectives (QHO), one for individual risk and one for societal risk. These QHOs are used as a basis for putting in context the computed risks from nuclear power plant operation. The QHOs are as follows.

"The prompt fatality QHO is that the risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 1/10 of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed. This represents a frequency of prompt fatalities of less than 5x10-7 per year for an average individual within 1 mile of a plant.

The cancer fatality QHO is that the risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes. This represents a frequency of cancer fatalities of less than 2x10-6 per year for an average individual within 10 miles of a plant."

These are the QHOs for nuclear power plant operation in the U.S. Compliance with these QHOs is viewed as a basis for establishing, by comparison to the numerical limits, that the risks posed by SFP operation are safe. These numerical QHOs are not viewed as hard limits, but risk results which are much lower than these limits provide a credible basis for concluding that the risks of nuclear plant operation are acceptable. Comparison against these numerical objectives requires an assessment of accident sequences involving SFP fuel uncovery events, assessment of the extent of radioactivity release from the fuel present in the SFP for each accident sequence, modeling the transport of the released radionuclides away from the SFP as they pass through the FHB to offsite, and then evaluating the public health impacts to the surrounding population. Because of the large uncertainties in such analytical assessments, sensitivity analyses are also performed to evaluate bounding or high health effect estimates by varying the modeling assumptions.

Such analyses of SFP fuel uncovery accidents have not been performed previously for the SFPs at DCPP. However, the USNRC has performed such analyses for four groups of representative plants operating within the U.S. (USNRC, 2013). The specific designs of the SFPs at DCPP were not chosen as one of the four representative designs. The similarities (i.e., PWRs with shared SFPs) are considered adequate for the highest group risks, Group 4, to bound the SFP risks of the DCPP.

SFP operation risk results for Group 4 plants are presented in Section 1.2 of Enclosure 1 of the USNRC study (USNRC, 2013a). That analysis concluded that an SFP fuel uncovery event would not result in any offsite early fatalities from acute radiation effects, so that the defined QHO for prompt fatalities (i.e., less than 5E-7 prompt fatalities per year) is then met. Further the USNRC's results showed that the risk of an individual within 10 miles of DCPP dying of cancer from the radioactivity released by an SFP fuel uncovery event would be less than 0.76% of the QHO for latent cancers (less than 0.76% of 2E-6 latent cancers per year).

Additional analysis details are provided in Enclosure 1 of the USNRC study (USNRC, 2013). The Group 4 bounding frequency evaluated for all sequences leading to damage to the fuel within the SFP (using the high estimates of the sensitivities that were performed) was 3.46E-5 per year. This total SFP fuel damage frequency includes all accident causes. The damage frequency is dominated by seismic events with acceleration levels well beyond design basis. The USNRC analysis also concluded that the conditional probability of an individual latent cancer fatality given the occurrence of an SFP fuel damage accident involving the Group 4 SFP design is very low (4.4E-4). Combining these two values results in an individual latent cancer fatality risk of 1.52E-8 per year, just 0.76% of the QHO for latent cancers. The USNRC study concludes that the conservative risk estimates of SFP operation for the plants represented by the Group 4 analysis bin is acceptable since it is well below the defined QHOs. The base case estimate for SFP fuel damage frequency for the same Group 4 plants is even lower (only 3.74E-6 per year), a factor of 9 still lower than the bounding frequency of SFP fuel damage events reported above. The study also performed extensive cost-benefit analysis and concluded that proposed design changes were not cost effective.

For these reasons, the risks from the operation of SFPs at DCPP are judged likely to be in compliance with the QHOs defined by the USNRC, so long as the frequency of SFP fuel damage is around the same SFP fuel damage frequency, 3.46E-5 per year. The results presented in Table 4-7 show that the SFP fuel damage frequency for DCPP is actually lower than 3.46E-5.

Many assumptions went into the USNRC Enclosure 1 (USNRC, 2013) risk results described above. The risk results are considered bounding as a result of conservative assumptions. Key modeling assumptions are noted.

1. A high estimate of total SFP cesium inventory of 142.2 MCi was assumed to be in the SFP (USNRC, 2018, Table 35). This is more than twice the single SFP cesium inventory for DCPP (USNRC, 2018, Table 72).
2. The SFP would not be coolable 100% of the time that fuel uncovery occurs for the Group 4 plants representative of DCPP (USNRC, 2013a, Table 42).
3. A non-mechanistic value of 90% of the cesium inventory present would be released given an SFP fire (USNRC, 2013a, Table 52).
4. Credit for evacuation of the local population within 10 miles of the plant was taken as documented (USNRC, 2013a, Table 63). Except for the small percentage of the population assumed not to evacuate (0.5%), all five of the other population cohorts were assumed to start the evacuation between 1 hour and 6 hours after being notified. There are a very small number of members of the public within 5 miles of the DCPP site.
5. None of the risk results reported above credit the added mitigation measures implemented in the U.S. for all plants with high-density SFP storage racks, such as are present at DCPP. These mitigation measures were implemented at DCPP following the Fukushima seismic and tsunamis events in Japan.

The SFP risks evaluated by the USNRC and described above consider the contribution from SFP operation per year assuming that there is no change in the number of fuel assemblies present in the SFP; i.e., as if no fuel assemblies are transported to the ISFSI for DCPP. These conditions are equivalent to a no fuel assembly offload scenario, which is clearly bounding for the SFP. For each of the four offload scenarios evaluated, fuel assemblies instead are to be transferred out of the SFP at DCPP to the ISFSI. This transfer of fuel assemblies lowers the total decay heat and cesium inventory in the SFP with time. Eventually, for each of the four offload scenarios, the SFP is emptied of fuel assemblies and the risk is then zero. To properly evaluate the difference in risks between offload scenarios, a risk measure that considers these time-dependent changes in the SFP inventories is required. The development of such a risk measure is described in Section 6. The risk measure provides a basis for comparing the risks of the offload scenarios by distinguishing the time-dependent amount of fuel that may overheat and release radiation in response to an event involving a loss of SFP cooling or coolant. The risk measure does not compare the risks between offload scenarios in terms of public health impacts. Instead, the work performed by the USNRC indicates that the risk of SFP operation for plants similar to DCPP is already safe. The conditional probability of an individual latent cancer fatality given the occurrence of an SFP fuel damage accident, evaluated using worst case modeling assumptions is judged applicable to the offload scenarios evaluated for DCPP. The risk measure applied here simply distinguishes the differences in risk between offload scenarios.

4 Screening of DCPP Fuel Handling Activities and Initiating Events

Past nuclear industry and U. S. Nuclear Regulatory Commission PRA studies on both boiling water reactor (BWR) and pressurized water reactor (PWR) plants, the DCPP type, have been reviewed to obtain insights as to the initiating events likely to contribute most to the risks from handling used nuclear fuel. More precisely, the initiators which are most likely to lead to differences in the comparative risks of large radiological releases from the four evaluated offload scenarios are of most interest. Some initiators may contribute to the risk posed by handling spent nuclear fuel, but generally contribute the same risk for all four offload scenarios. These need not be evaluated for the purpose of this study.

A comprehensive list of potential initiating events affecting the fuel handling at DCPP is provided in Table 4-1. This list of initiating events is for consideration in the current risk assessment. They may have been quantified in SFP risk studies for other plants, or considered for other plants and screened out, but such screening may not necessarily be applicable to DCPP. This section and the associated tables in Appendix A document the judgments regarding screening of the events listed below for applicability to DCPP. Table A-1 presents the screening of potential initiating events for fuel handling and storage inside the FHB. Table A-2 presents the screening of potential initiating events which affect fuel handling and storage outside the FHB The frequencies of initiating events leading to potential fuel damage are quantified and summarized in Table 4-5.

Table 4-1. Potential Initiating Events

Groups of Initiating Events	Potential Initiating Events in Group
A. Heavy Load and Fuel Assembly Drops	1. Drop of an MPC transfer cask (while empty) onto (a) the cask pit; (b) SFP; (c) concrete floor. 2. Cask tipover. 3. Drop of an MPC onto the storage overpack. 4. Drop of the transfer cask loaded with MPC onto (a) the cask pit; (b) SFP; (c) concrete, asphalt, or gravel during transport. 5. Tipover of the transfer cask with MPC (a) while cask moved; (b) when impacted by vehicle; (c) when stationary. 6. Loss of transfer cask supplemental cooling system. 7. Drop or tipover of the VCT loaded with (a) transfer cask, (b) storage overpack. 8. Drop of a fuel assembly while being moved under water in the SFP.
B. Losses of SFP Inventory	1. Equipment/tools falling into SFP puncturing liner. 2. Other random liner leaks at varied elevations. 3. Breaks of SFP connecting piping to SFP. 4. Loss of inventory through transfer canal when transfer basin is full and not isolated. 5. Siphoning of SFP inventory through SFP cooling system, makeup piping, or skimmer pump system. 6. SFP gate or seal failure, or loss of compressed air. 7. SFP cooling pump leaks leading to inventory loss. 8. Skimmer pump leaks leading to inventory loss. 9. Refueling water purification pump leaks leading to inventory loss.
C. Losses of SFP Cooling	1. SFP pump fails to run. 2. SFP heat exchanger clogs. 3. SFP filter clogs. 4. Loss of surge tanks for secondary heat removal. 5. Loss of supporting systems for SFP cooling: (a) AC vital power; (b) losses of offsite power; (c) component cooling water; (d) auxiliary saltwater.
D. Spatially Dependent Internal Events	1. Internal fires (e.g., in auxiliary building or SFP area). 2. Internal floods, including by spent fuel pool leakage.
E. External Events (Resulting in Impacts, Losses of Inventory, or Losses of SFP Cooling	1. Seismic. 2. Aircraft crashes. 3. Meteorite impact. 4. Explosions. 5. Pipeline explosions. 6. Truck or railcar explosions. 7. External flooding. 8. External fires. 9. High winds. 10. Tornado missiles. 11. Turbine missiles. 12. Security /sabotage/terrorism events. 13. Onsite or nearby chemical releases. 14. Extreme external temperatures. 15. Ocean shipping.
F. Criticality Events	1. Miscalibration of SFP boron concentration; 2. Reduction of boron absorber in fuel rack panels; 3. Loading of an unauthorized fuel assembly into an MPC; 4. During transfer or storage of fully loaded MPC as ISFSI.
G. Loss of Heat Removal from Storage Overpack Installed at ISFSI	1. Blockage of ventilation ducts including by landslides. 2. Adiabatic heatup.

The potential initiating events in Table 4-1 are organized into groups labeled A, B, C, D, E, F, and G. The initiating events listed in Table 4-1 derive from past risk studies of spent fuel pools, from the Updated Final Safety Analysis Report for DCPP (PG&E, 2018) and the ISFSI UFSAR (PG&E, 2018a) which evaluated the risks for all design basis events.

Table 4-1 provides a starting point for potential generic initiating events that may be applicable to DCPP. Their applicability and risk significance to DCPP is presented in Appendix A, Tables A-1 and A-2. The assessment in Appendix A is a further screening and quantification process to examine which initiating events are to be evaluated for this risk assessment.

Before describing the assessment of each group of potential events as applicable to DCPP, Table 4-2 lists fuel uncovery event frequency results for some of the initiating events developed in past studies performed by the USNRC and its contractors (USNRC, 1987), (USNRC, 1989) and (USNRC, 2001). Among the three USNRC studies available, there is at least one initiating event quantitatively evaluated from each of the major initiator Groups A, B, C, D and E, but not for Group F or Group G (criticality events or losses of heat removal from the ISFSI storage overpacks).

Table 4-2. Comparison of PWR SFP Fuel Uncovery Frequencies (events per SFP-year)

Initiating Event	**NUREG/CR-4982 (1987)**	**NUREG-1353 (1989)**	**NUREG-1738 (2001)**
A. SFP structural failure from cask drop	<2E-8	3.10E-08	2.00E-07
B. Loss of SFP water inventory	NA	4.20E-08	NA
C. Loss of pool cooling or makeup	5.70E-07	6.00E-08	1.70E-08
C. Loss of offsite power	NA	NA	1.40E-07
C. Loss of pool water due to pneumatic seal failure	5.00E-07	3.00E-08	NA
D. Internal fires	NA	NA	2.30E-08
E. Seismic structural failure of SFP	5.0E-6 (77%)	1.80E-6 (91%)	2.0E-6 (83%)
E. Structural failure from aircraft crash	<1E-10	6.00E-09	2.90E-09
E. Structural failure from turbine missiles	4.00E-07	1.00E-08	<1E-9
F. Criticality events	NA	NA	NA
G. Loss of heat removal for storage overpacks installed at the ISFSI	**NA**	**NA**	**NA**
Total	**6.47E-06**	**1.98E-06**	**2.40E-06**

NA – indicates that the initiating event class was either not assessed or assessed to not apply and therefore has zero frequency contribution.

For initiating events listed under Group A in Table 4-1, a transfer cask with a fully loaded MPC drop into the SFP is selected for assessment in this study (A.4.c.) For completeness, the drop of the empty MPC loaded in a transfer cask, when it is first being transferred into the SFP for the loading of fuel assemblies is also selected for assessment (again A.4.c.) After fully loading an MPC inside a transfer cask with fuel assemblies, the transfer cask and MPC are raised above the SFP water line for decontamination prior to transport to dry storage (about 40 feet above the SFP floor). A drop of the transfer cask from this height may potentially fail the SFP liner and concrete structure if it lands on the SFP floor, or to a lesser extent, if it impacts the top of the SFP wall. Such a drop may conceivably open a pathway for draining the SFP water. If the size of the opening is large enough, there may then be insufficient time to provide makeup to the pool to restore level and maintain water above the fuel assemblies.

Analyses were performed by PG&E for the DCPP specific design of the transfer cask system and SFP liner and concrete structure. These analyses were accepted by the USNRC although they concluded otherwise; i.e., the SFP would not develop a leak, even from the greatest fall height feasible. See Section 9.1.2.3.11 of Chapter 9 of the DCPP FSAR (PG&E, 2018) and (PG&E, 2019f). The analysis concluded that the liner would be perforated, that the concrete slab would only be penetrated about 0.6 inches, and that the rock foundation would absorb the energy of impact with an elastic deformation estimated at only 1.12 inches. Therefore, the conclusion was that there would be no perforation through the slab allowing the SFP coolant to leak.

The U.S. Nuclear Regulatory Commission (USNRC, 2014) also evaluated an SFP structure from such a heavy load drop at a different PWR nuclear power plant and reached the same conclusion. However, the USNRC still assumed that the conditional probability of a leak developing given the drop occurred would be 0.1. This probability is viewed as conservative given the DCPP specific analysis but is included here for the basis of comparing risks from other initiating events. It will be shown later that even assuming a 0.1 conditional probability of an SFP leak given a drop that risks from such events are relatively small.

Fuel handling events at locations away from the SFP, but still inside the FHB, and for handling activities outside the FHB are expected to have approximately the same frequencies and release consequences regardless of the offload scenario chosen. For this reason, fuel handling events away from the SFP and outside the fuel handling building are not considered further. Accidents involving these activities outside the FHB would lead to consequences limited by the number of fuel assemblies and maximum allowed heat load within the single affected MPC; i.e., far fewer fuel assemblies and heat load than for the transfer cask drop into the SFP selected for this assessment, where all fuel assemblies in the SFP are potentially at risk. A summary of potential initiating events caused by fuel handling activities within the FHB is presented in Table A-1 of Appendix A. A summary of potential fuel handling events outside the FHB and from external initiating events there and at the ISFSI is presented in Table A-2 of Appendix A and highlights from that assessment are discussed below. The insights documented in Tables A-1 and A-2 come from a variety of sources both generic and DCPP specific (PG&E, 2018; PG&E, 2018a). The specific reference(s) for each entry are presented in the tables.

Fuel assembly drops within the SFP are considered adequately addressed by design. The risks are negligible compared with other postulated severe accident progressions. (EPRI, 2014). There have been at least 11 events in which fuel assemblies have been dropped during a fuel assembly movement. None of the 11 dropped fuel assemblies resulted in fuel damage (USNRC, 2007).

For potential initiating events listed under Group B in Table 4-1, the same drop of a transfer cask with a fully loaded MPC into the SFP (A.4.c in Table 4-1) may also result in a loss of SFP cooling. It is conservatively assumed to result in a loss of SFP coolant inventory with a conditional probability of

0.1. Once the SFP coolant level drops about 4 feet below the normal water level, the SFP cooling suction line is uncovered. SFP makeup water is then required before SFP cooling can be restored.

Item B.6 in Table 4-1 considers failure of the pneumatic seal between the main SFP and the transfer canal. Such a failure could conceivably transfer SFP water inventory rapidly to the transfer canal depending on the size of the gap to be sealed. The inflatable gate seal is normally pressurized by instrument air and at DCPP this pressure source is backed up by bottled nitrogen. These nitrogen bottles are design Class II (non-seismically qualified). The weir at the bottom of the gate is located just above the top of the fuel assemblies. While a loss of SFP cooling would likely result, there would still be ample time to provide SFP makeup and align for SFP coolant using the variety of methods available before coolant boiling begins. If there are no refueling operations in progress, the fuel transfer tube is isolated from the transfer canal. If the gate seal were to fail when the transfer tube is isolated, the SFP water level would equilibrate at an SFP water level much higher than at the bottom of the weir wall; i.e., just 3 feet lower than the normal SFP coolant level (PG&E, 2019c). Therefore, the potential failure of the pneumatic seal was excluded from further consideration as an initiating event.

Other random losses of SFP coolant inventory identified in Group B of Table 4-1 are judged to have much lower impacts on the SFP. The liner leakage sizes would be lower and most likely the opening would be located higher in the SFP resulting in lower leakage rates, and hence would be less challenging to restore before fuel uncovery occurs. For these reasons, no other Group B initiating events are described in Appendix A.

For initiating events listed under Group C in Table 4-1 (losses of spent fuel pool cooling), in the absence of a concurrent leak, there would be ample time to restore cooling depending on the failure cause, or to align for alternate sources of SFP water makeup. Without any liner leakage, even soon

after a full core offload (193 fuel assemblies at DCPP), there would be more than 6 hours before the SFP water begins to boil as noted in Table F-2 of a risk study for PWRs performed by EPRI (EPRI, 2014). For DCPP SFPs, conservative analyses have estimated that water may begin boiling or evaporating rapidly within 4 hours.

In Group C of Table 4-1, a loss of offsite power at DCPP would initially shed the SFP cooling pumps. They must be manually restarted and loaded onto the emergency diesel generators to restore SFP cooling if the offsite power outage continues. Since the generic plant 2001 study (USNRC, 2001) was completed, additional procedures and equipment have been installed and portable equipment positioned to provide alternate means of SFP cooling and, if necessary, SFP injection in the event of a loss of all offsite and onsite emergency AC power. The DCPP specific FLEX procedure (PG&E, 2019), for example, describes the steps needed to align for alternate SFP makeup and for SFP cooling. These enhancements, implemented at all plants in the U.S., are expected to further reduce the frequency of SFP fuel uncovery events going forward.

For the above reasons, random losses of offsite power (other than those caused by external influences such as seismic events) and all other causes of losing SFP cooling are not considered It has been postulated that the VCT could come in contact with a power line which crosses the VCT road during movement of a transfer cask to the ISFSI. However, maximum height of the VCT lifting beam is administratively controlled to limit it below the approximately 55 feet of the Unit 2 500-kV transmission lines above the road surface.

Also at DCPP, filter clogging of the SFP cooling loop does not cause a loss of SFP cooling because the cooling path is in a different loop. While a seismic event may lead to failure of the component cooling water (CCW) surge tanks and a loss of normal SFP cooling, the sequence does not affect the line-ups available for gravity makeup and evaporative cooling to the SFP. Seismic initiated losses of offsite power (from Group E) are still considered along with the other equipment and structure failure impacts caused by seismic initiating events. Therefore, no initiating events from Group C of Table 4-1 were selected for quantification in this study. For the list of potential initiating events in Group C, only insights regarding the loss of offsite power event (Item C5.b in Table 4-1) are presented in Appendix A.

For initiating events listed under Group D (spatially dependent internal events), both internal plant fires and internal plant floods could potentially impact the SFP systems. Table 4-3 presents the results of analyses performed by PG&E on the DCPP for reactor at-power conditions (summarized from Table 4-1 of the DCPP risk study (PG&E, 2019a). The Unit 1 and Unit 2 results are similar. Internal fires contribute about 50% of the total core damage frequency (CDF) and 19% of the total large, early release frequency (LERF). The contribution from internal flooding events is about 10% to CDF and 3.5% to LERF. For internal fires, a large part of the contribution is from fires initiating in the turbine building, where the emergency switchgear is located in cubicles separated by divisions. For internal floods, firewater pipe ruptures contribute significantly on a percentage basis for reactor at-power conditions but only about (8E-6/year to CDF and only 3E-7/year to LERF). A firewater pipe rupture in either flood zone 31 or 32 could potentially, if not isolated, flood the auxiliary feedwater (AFW) pumps and residual heat removal (RHR) on the associated reactor unit. Such an event could possibly also affect SFP cooling although a detailed assessment of the impacts on SFP cooling systems from potential flooding in these areas has not yet been performed.

For this study, it is judged that such internal fires or internal floods would likely only disable SFP cooling and potentially preclude access to the SFP cooling pump rooms for a time. Such an impact on initial SFP cooling receiver actions would still allow ample time for manual actions to align for SFP makeup prior to SFP fuel uncovery. Therefore, no initiating event from Group D is selected for evaluation in this study. Also, for these reasons, no other Group D potential initiating events are described in Appendix A. The accessibility to SFP areas needed to provide SFP makeup under internal fire or internal flood conditions should be considered further.

For potential initiating events under Group E (external events) in Table 4-1, the list of possible initiating events is long. Appendix A documents the basis for screening most external initiating events for this study. Seismic events, on the other hand, were found to contribute substantially (roughly 65% to both CDF and LERF) for reactor at-power CDF and LERF at DCPP (see Table 4-3). For DCPP the seismic hazard frequencies for the corresponding accelerations are considerably higher than for east coast PWRs, as noted in the DCPP updated FSAR (PG&E, 2018). Based on past SFP evaluations for other nuclear plant sites, and the higher seismic hazard applicable to DCPP than was used in these evaluations, one might expect the risks from seismic events for the DCPP SFPs to be greater than 3E-6 per SFP-year at DCPP. However, DCPP has also been designed for higher seismic accelerations (a double design basis earthquake of 1.75 g SA and for the Hosgri fault up to 2.1 g SA). The risks from seismic events must therefore consider a combination of the hazard curve frequencies and the associated seismic capacities of systems, structures, and components available at DCPP.

The contributions listed in Table 4-2 from seismic events assessed for other plants leading to structural failure of the SFP are found to dominate the total frequencies of SFP fuel uncovery in each of three USNRC references (USNRC, 1987; USNRC, 1989; and USNRC, 2001). They contribute 80 to 90% of the total SFP fuel uncovery frequency. These results were all obtained prior to more recently assessed, and now typically higher, seismic hazard frequencies for the U.S., and were performed only for plants in the eastern U.S. The seismic hazard frequency for DCPP and for other sites in the western U.S. are higher than for any of the nuclear plant sites previously assessed.

Based on the above, seismic events impacting the DCPP SFP and its cooling systems are selected for detailed assessment. No other external initiating events impacting the SFP and its cooling systems were selected for study. Seismic events are also selected for assessment of the impacts on the ISFSI storage overpacks. Seismic initiated rock sliding onto the ISFSI storage pads was also considered but screened out as not significant. Much work has been performed previously on this subject by PG&E and is referenced in Appendix A.

A small number of other external initiating events were also selected for study of their potential impacts on the ISFSI storage overpacks. The initiating event impacts come only from events that are well beyond the design basis already considered in licensing analyses. As expected, these are low frequency events. Appendix A documents the insights of this screening process and the basis for associated sequence frequencies which were evaluated. Table 4-5 summarizes the quantitative results for initiating events which may impact either the SFP or the ISFSI storage overpacks.

For potential initiating events under Group F (criticality events) in Table 4-1, the list of possible initiating events is short. Past studies performed by the USNRC and its contractors considered criticality events (USNRC, 1987), (USNRC, 1989) and (USNRC, 2001). Each of these studies' authors judged that the risk of inadvertent criticality events in the SFP leading to fuel uncovery is insignificant. Thus, a formal quantification of criticality events is considered unnecessary based on the physical and procedural measures to maintain subcriticality in the SFP.

At DCPP the SFP coolant is maintained with 2000 parts per million of boron (minimum) for reactivity control. Boraflex is not used to control reactivity within the SFP and has been largely removed. In response to an accident requiring makeup to the SFP, both borated and unborated water sources are available. The borated sources are preferred. If the makeup water is not borated the boron concentration can be conserved by adding an amount of unborated water equal to the amount evaporated. If the SFP coolant boron concentration drops to 800 ppm, k_{eff} (an abbreviation for the reactivity coefficient, k-effective) it is still less than 0.95 (PG&E, 2018). Also, in Section 9.1.2.3.10 of that same reference, for DCPP even if the SFP is filled with unborated water there is a "95% probability at 95% confidence that k_{eff} is still less than 1.0". However, at DCPP makeup to the SFP regardless of the source is to be completed prior to coolant level lowering to below the top of the fuel. Further, the high-density fuel racks within the SFP have been qualified to the Hosgri level of 2.1 g SA. While the behavior of the fuel racks at still higher accelerations has not been studied, the grid structure is made up of tube steels so that they can slide. No blockage of water passages within the fuel assemblies is expected.

The DCPP licensing requirements also call for still higher boron concentrations in the SFP during cask loading operations to account for the cask specific configuration. The loss of boron concentration as a result of SFP coolant boil-off or drainage of coolant is still not expected to result in a reactivity concern following a large seismic event. This conclusion is supported by the USNRC (USNRC, 2013, Appendix G). In response to a related public question, the USNRC noted the following after describing the design measures used to prevent such an inadvertent criticality.

"Since these measures may be challenged by a beyond design-basis event, the NRC staff cannot rule out the potential for an inadvertent criticality event. However, the NRC staff judges that the potential consequences of a zirconium fire in the SFP and an associated hydrogen deflagration considered in this analysis would not be significantly affected by an inadvertent criticality event. The NRC staff bases this judgment on the following considerations.

1. While the earthquakes considered in this analysis are beyond what the fuel was designed to withstand, it is not likely that the fuel would experience sufficient damage to cause significant changes in the geometric configuration of the fuel needed to cause inadvertent criticality.
2. The necessary moderator would tend to shield and contain the effects of a criticality such that it would primarily pose an onsite rather than offsite hazard.
3. Criticality requires the presence of a moderator and therefore power would not be sustained as the pool lost inventory due to boiling or drain down. Since the power generated by any inadvertent criticality would be far lower than in the reactor, the inadvertent criticality would have negligible impact on the long-lived fission product isotope inventory. The additional short-lived isotope inventory would not result in any early fatality risk because of the emergency response as modeled precludes such exposure. This is due in part because of the length of time needed before any fission products are released offsite. Therefore, any offsite release associated with a criticality would be small relative to potential releases from a zirconium fire."

Fuel assemblies are dry while loaded in an MPC during transit from the SFP to the ISFSI, or while at the ISFSI in storage overpacks. The MPCs are pressurized with helium. The ISFSI FSAR (USNRC, 2016, page 4.2-26) indicates that with the storage overpack dry and hence no moderator, the reactivity is very low (K_{eff} less than 0.515). This dry configuration makes it very difficult for any criticality event to occur. Therefore, such criticality events are not evaluated further. Thus, no Group F potential initiating events are described in Appendix A.

Potential initiating events listed under Group G (loss of heat removal from storage overpacks at the ISFSI) in Table 4-2 are considered here and in Appendix A, Table A-2. The passive nature of heat removal from the ISFSI storage overpacks is very reliable. Only extreme beyond design basis events have the potential to cause loss of heat removal capability. Such events are seismic, tornados, and aircraft crashes listed in Table 4-1 and Appendix A. These events have the potential to result in the loss of heat removal to one or more ISFSI storage overpacks. Table 4-5 summarizes the frequency results from the assessment of Group G initiating events which survive the screening as documented further in Appendix A.

One other potential initiating event from Group G is a rockslide involving slopes above or below the ISFSI. The issue is a landslide could result in the loss of natural convection, that is, the passive cooling for the stored fuel. Of course, some heat transfer would still occur via conduction. This potential initiating event is mentioned because of the massive slide that occurred in early 2017 at California's Big Sur Pacific Coast highway. The hillside above the highway gave way after several days of heavy winter and spring rains. An assessment (Geo Engineer, 2019) attributed the Big Sur landslide to the heavy rains. While the slope was already known to be creeping downhill at the rate of 17 cm per year, the added rain led to water replacing air between soil particles causing the hillside stresses to change, destabilizing the slope. An assessment of the applicability of this type of event to the ISFSI is described.

The ISFSI FSAR (PG&E, 2018a, Section 2.6) section on "Geology and Seismology" makes it clear that "There are no active landslides or other evidence of existing ground instability at the ISFSI or CTF sites , or on the hillslope above the ISFSI site." Nevertheless, the stability of the hillslope and the slopes associated with the pads, CTF, and transport route was analyzed in detail using conservative assumptions. The analysis showed there were ample factors of safety to avoid a landslide that could impact the storage overpacks at the ISFSI. A key difference with the situation at Big Sur is that the ISFSI is sited on bedrock.

The stability of the hillslope above the ISFSI was analyzed for the case when the clay beds are saturated with water (PG&E, 2018a, Section 2.6.5.1). The results showed that the hillslope is stable with suitable margins of safety (PG&E, 2018a, Section 2.6.5.1.2.4). The USNRC reached the same conclusion for stability using an alternate analysis approach. Therefore, a landslide caused by heavy rains such as occurred at Big Sur is ruled out.

The ISFSI FSAR analysis also evaluated the occurrence of a large seismic event that might lead to a landslide. An ISFSI long-period earthquake (ILP) spectra was developed for this seismic evaluation. The ILP spectra was then used for the design analysis and it was shown that any such landslide would be limited to 3 feet of sliding. The setback distance between the edge of the ISFSI pads and the toe of the cut slope above the pads is about 30 feet. Additional questions are how much stronger would a seismic event have to be to impact the ISFSI and what would be the frequency of such an event. PG&E prepared a report to answer these and related questions (PG&E, 2020). The results are briefly summarized.

The mean hazard curve shows that for a 1.8 g SA at 5 Hz as analyzed for the ILP spectra, the exceedance frequency of such seismic events is 2E-4 per year. The extended analysis assumed that the landslide would have to slide 15 feet (half the actual setback distance) to allow for the possibility of portions of the individual segments moving the full 30 feet available. A seismic event with 9.8 g SA at 5 Hz was determined to be needed to cause a slide distance of 15 feet. The available seismic hazard exceedance curve frequency was then evaluated at 9.8 g SA to arrive at a frequency just less than 1E-7 per year. The mean hazard exceedance curve was extrapolated from 6.0 g SA to 9.8 g SA. The median hazard exceedance curve does extend to 1E-7 per year at 4.0 g SA.

The results from this study (PG&E, 2020) are to be interpreted as a bounding analysis due to the conservative assumptions made to simplify the analytical models, although the assumed arrangement and continuity of the slope clay beds were chosen to maximize the amount of slope displacement calculated. Nine conservative assumptions listed in the PG&E report were thereby not fully captured in the analysis models used.

It is instructive to compare the results by initiating events from past evaluations of SFP risks (see Table 4-2). These results come from the USNRC's evaluation in support of Generic Safety Issue 82 and SFP accident risk for decommissioning (USNRC, 1987; USNRC, 1989; and USNRC, 2001). After seismic events, the risks from losses of SFP cooling or makeup and failure of the pneumatic gate seals within the SFP contribute the next highest frequencies in the earliest study from 1987. The latter two studies in 1989 and 2001 indicated these initiating events contribute much less to the frequency of SFP fuel uncovery. By contrast, the USNRC's study of SFP risks for decommissioning (USNRC, 2001) evaluated frequencies from cask drops into the SFP and a loss of offsite power each contribute about

10% of the seismic initiated frequency of SFP fuel uncovery. In this assessment for DCPP, a loss of offsite power is assumed for all seismic events modeled.

Table 4-3. Total DCPP CDF/LERF for Plant Internal Events, Seismic Events, Internal Fires, and Internal Floods for At-Power Conditions (events/year)

Initiator categories	CDF	% CDF U1	%CDF U2	LERF	% LERF U1	%LERF U2
Internal Events (either unit)	4.98E-06	5.62%	6.11%	1.01E-06	12.53%	12.78%
Seismic events (either unit)	2.83E-05	31.95%	34.72%	5.17E-06	64.14%	65.44%
Internal Fires Unit 1	4.63E-05	52.28%		1.58E-06	19.60%	
Internal Fires Unit 2	4.01E-05		49.20%	1.45E-06		18.35%
Internal flooding U1	8.99E-06	10.15%		3.00E-07	3.72%	
Internal flooding U2	8.13E-06		9.97%	2.72E-07		3.44%
Total for Unit 1	8.86E-05	100.00%		8.06E-06	100.00%	
Total for Unit 2	8.15E-05		100.00%	7.90E-06		100.00%

In 2013 and 2014 the USNRC (USNRC, 2013 and USNRC, 2014) also reported the results of a more encompassing study of the seismic risk of SFP fuel uncovery for different groups of nuclear plants of similar designs, all located in the eastern U.S. The results for their defined Group 2 category of nuclear plant designs for PWR plants similar in design to DCPP were found to have a seismic SFP fuel uncovery point estimate frequency of 3.28E-6 per SFP-year. The result for seismic events was again found to be about 90% of the total SFP fuel uncovery frequency.

Therefore, seismic events found to be important contributors for DCPP at-power conditions and at other plants for SFP operation are selected for the evaluation of SFP fuel uncovery frequency for DCPP. The different fuel offload scenarios are not expected to have different frequencies of direct seismic event impacts leading to SFP fuel uncovery. However, the radiological consequences of such events would depend on the decay heat, age, and number of fuel assemblies present in the SFP at the time of the seismic event. The number of fuel assemblies in turn depends on the fuel offload scenarios being evaluated.

PG&E has also evaluated the other external events in Group E of Table 4-1 for applicability to DCPP at-power operation. Attachment 1 of the DCPP PRA for external events during at-power conditions (PG&E, 2016a) summarizes the screening of those external events not applicable to DCPP and the results of those evaluated quantitatively. Bounding quantitative results were used to show that the frequencies of the following initiating events leading to core damage were each less than 1E-6/year and hence were screened from further evaluation: aircraft crashes, external flooding, tsunamis, extreme winds or tornados, and transportation accidents. More is discussed about these external events and their applicability to spent fuel risk in Table A-2 of Appendix A.

While each offload scenario must offload the same total number of fuel assemblies, and therefore must offload roughly the same total number of loaded MPCs to the ISFSI, the time at which they occur and the number of fuel assemblies present in the SFP at the time of a cask drop differs among the four offload scenarios to be evaluated. It is noted that the offload scenario titled the "Pre-Shutdown Vendor Option Offload" calls for a slightly larger MPC that can hold up to 37 fuel assemblies, but this difference only leads to a reduction from 40 to 35 of fully loaded MPCs to be transferred to dry storage for each unit.

Table 4-4 compares the results of seismic events to all other initiating events for an SFP risk generic study for PWRs (EPRI, 2014) performed by the Electric Power Research Institute (EPRI). The assessment was performed as a pilot project for a PWR plant located on the east coast. A new focus of this study was to consider the dependencies between seismic events and their impacts on both the SFP systems and the reactor systems while the reactor is still producing power. DCPP Unit 1 is to operate until 2024 and DCPP Unit 2 until 2025. This most recent SFP PWR pilot study concluded that the frequency of SFP fuel uncovery, labeled as fuel damage frequency (FDF) in Table 4-4 was again dominated by direct seismic impacts on the SFP structure. In actuality, the loss of coolant sufficient to uncover the fuel within the fuel assemblies would not necessarily lead to fuel damage. A discussion of radioactivity release given spent fuel uncovery is presented in Subsection 8.4.

Table 4-4. Contributors to PWR Pilot SFP Fuel Uncovery Frequencies (per SFP year), (EPRI, 2014)

Contributors to SFP Metrics	FDF	LERF
Seismic Structural Failure of SFP	7.72E-07	7.68E-07
Seismic At-Power Impacting SFP	1.46E-07	1.46E-07
All Other	5.97E-10	5.43E-10
Total	**9.19E-07**	**9.15E-07**

Sixteen percent of the total SFP fuel uncovery frequency was contributed by seismically initiated sequences causing a severe accident involving the operating reactor, and as a result indirectly leading to the SFP fuel being uncovered. Consideration was given to hydrogen produced by a severe accident in the operating reactor then leaking into the FHB where the SFP is located. There was a concern following the March 11, 2011, earthquake and tsunamis at Fukushima, Japan, that such hydrogen could potentially ignite near the adjacent SFPs leading to a loss of SFP cooling and eventual SFP fuel uncovery. While hydrogen explosions migrating from other locations did occur in the Unit 4 SFP building during the Fukushima accident, the impacts of this explosion did not cause the fuel within the SFP to uncover. These events were evaluated in the EPRI PWR pilot study (EPRI, 2014). The results were significant enough to conclude that such indirect seismic events should also be considered for this DCPP assessment. DCPP has relatively high seismic hazard frequencies compared to corresponding accelerations at the EPRI assessed pilot plant located in the eastern U.S.

The tsunami triggered by the earthquake near Fukushima caused nearly all the plant damage. A PG&E study of external events (PG&E, 2016a) assessed the frequency of tsunami events of sufficient wave height to affect reactor operation at DCPP as being very low. The auxiliary saltwater system is potentially affected for wave heights as low as 20 feet mean lower water level (MLWL) if the watertight pump room doors are not fully closed. Access to the auxiliary saltwater (ASW) pump rooms is monitored by the control room personnel and an alarm will sound in the control room if any of the watertight doors are open making it unlikely that the doors will be left open. If the watertight doors are closed, as they should be, then the ASW pumps could still be impacted at a height of 48 feet MLWL due to possible ingestion through the ASW system snorkels. The assessed frequency of exceeding 48 feet MLWL is only approximately 8E-7 per year (PG&E, 2016a). The frequency of a tsunami directly affecting operation of the SFP whose floor is located at the 99-foot elevation in the fuel handling building at DCPP is much lower. SFP cooling depends on CCW and it, in turn, depends on the ASW system for its ultimate heat sink for normal SFP cooling. Given loss of SFP cooling, however, there are ample systems available for makeup to the SFP. The direct impact of seismic events on the auxiliary saltwater system and AC power is much greater than from any accompanying tsunami. Further, the dry storage casks at the ISFSI are located on a hill about 200 feet above the SFP floor elevation. So dry storage at the ISFSI would also not be affected by tsunami events. Therefore, the direct impact of a tsunami on either dry storage or SFP operation is not evaluated further.

DCPP currently has two operating reactors and two separated SFPs. Until both SFPs are emptied of fuel assemblies, the frequencies of SFP fuel uncovery events caused by dropping a transfer cask into an SFP can be considered doubled that of one SFP since about 40 MPCs are yet to be offloaded from each unit's SFP. This is instead assessed as evaluating the risks of a transfer cask drop in the SFP for Unit 2, but then doubling the frequency of fuel uncovery assuming the risk from Unit 1 SFP is similar. However, the consequences of such a transfer cask drop event into the SFP that also results in a coolant leak would be limited to, at most, the fuel assemblies in the affected SFP at the time. The time-dependent nature of the number and decay periods of the fuel assemblies in the Unit 2 SFP with time is accounted for.

For seismic events, the frequency of an SFP fuel uncovery caused by structural failure of the SFP itself (involving one unit or the other, or both at the same time if both contain spent fuel) is evaluated to be that of a single Unit 2 SFP. A common approach to seismic risk assessment is to correlate the seismic failures of identical structures and equipment at the same site subject to the same shaking environment. The response of the two SFPs to seismic shaking is therefore judged to be highly correlated so that if one unit's SFP structure fails, they both are assumed to fail. Performing the risk assessment only for the SFP of Unit 2 simplifies the analysis.

In response to a strong seismic event the set of impacts and the progression of events proceed similarly. Both SFPs may undergo an SFP fuel uncovery at the same time. The extent of fuel damage by the SFP uncovery events would be added. Of course, this applies only if both SFPs still contain fuel assemblies.

Unit 1 refueling outages are staggered so that Unit 1 outages generally follow those for Unit 2 by 8 to 12 months. Unit 1 will be permanently shut down and fully offloaded of spent fuel to the SFP 10 months earlier than Unit 2. Therefore, when Unit 2 undergoes refueling, the Unit 1 has experienced 8 to 12 months cooling of the most recently transferred fuel since its most recent refueling. When the decay heat load in Unit 2 is highest, the most recent fuel offloaded to Unit 1 has cooled for 8 to 12 months. The total time between Unit 2 refuelings varies but is typically 19 to 21 months, except for the final shortened refueling interval. The reactor permanent shutdown of Unit 1 follows the most recent refueling of Unit 2 by about 1 month. This means that the periods of greatest total SFP heat load (in the first 30 days after reactor shutdown) in each unit do not overlap.

In response to a strong seismic event, the conditional probability of fuel uncovery due to a failure of the SFP structures may be nearly the same for both units. However, the extent of fuel damaged in the unit with a longer cooling time since experiencing a reactor offload should be less. The unit with the longer SFP cooling time changes during the fuel cycle time, because the units alternate refueling outages. Obviously, when Unit 1 is fully emptied but the SFP for Unit 2 still has fuel assemblies, the added contribution from Unit 1 to the total amount of fuel damage would be zero. After both reactors are permanently shut down in 2025, both SFPs will contain fuel assemblies until all the fuel assemblies for each unit are transferred to the ISFSI. During these years, the spent fuel assemblies in the Unit 1 SFP will have about 10 months of added cooling time as compared to those in the Unit 2 SFP. Any added contribution to the amount of fuel damaged from Unit 1 is not investigated further for consequence assessment, since scaling the results in whatever manner is applied would not significantly impact the risk rankings of the four offload scenarios being assessed.

With respect to the "progression of accidents", consideration has first to be given to the possible threats, or initiating events, to the integrity of the spent fuel followed by assessing what happens as a result of such threats. The risk assessment technologies most lacking are methods for evaluating the amount of radiological release from the SFP, and thereby the consequences of offloading spent fuel accidents.

The screening and judgments of applicability to DCPP of the potential initiating events listed in Table 4-1 are documented in Appendix A and the above discussions. The position is taken that two events bound all others in terms of the potential for a severe accident involving the storage and offloading of

SFP to the ISFSI. These two are accidents initiated by a severe earthquake, including those beyond the magnitude of the design basis earthquake and a dropped transfer cask (~117 tons) over the SFP. Both of these initiating events may directly impact the structures and equipment at the location of the highest concentration of actively cooled spent fuel. Their selection is very likely to govern the risk of spent fuel handling and transport at DCPP and is consistent with the findings of similar studies on other nuclear power plants. Except for a major sabotage or terrorism event, these are believed to be the only threats that could lead to a catastrophic partial or complete loss of water from the SFP. In the absence of the extensive number of safeguards performing their intended functions (discussed later), a partially or fully uncovered spent fuel pool has the greatest potential for a severe radiological release event associated with the storage and transport of spent fuel. Because these two initiating events are believed to be dominant with respect to a severe radiological release event, this is a surrogate of the total risk of the entire offload process, although consideration is given to the other phases of the offload scenarios, including the ISFSI.

Table 4-5 summarizes the frequencies for and identifies the initiating events that survived the screening documented in Appendix A. Seismic events (IDs 8 and 9) which may impact the SFP are also to be retained, but the frequency varies depending on the seismic magnitude. This complex hazard is discussed separately in Section 8.

All initiating event frequencies are presented “per year” when spent fuel is located in the SFP. For initiating events that are “demand-based”, in that they can only occur for discrete handling activities, the total probability for all such similar activities is summed and the total divided by 10 years as an approximation to the length of time for the calendar years studied—the time from the beginning of refueling outage 2R22 until all fuel is emptied from the Unit 2 SFP by transfer to the ISFSI. This is an approximation in that the actual time duration for fuel in the SFP varies for each offload scenario between 9 and 13 years, as noted in Section 8.1, Table 8-2. By this scaling to per-year, the “demand-based” events are also assigned an effective event frequency per year, independent of the specific offload scenario. This enables their frequencies to be straightforwardly compared to the frequency-based events expressed in events per year. The vendor option has the longest duration (12.7 years) but it is worth noting that the small number of fuel assemblies late in the study period is likely not vulnerable to fuel overheating.

Table 4-6 presents sums of the Table 4-5 initiating event frequencies per year by groups of potential consequences. Sum-1 is for the total frequency of randomly dropping a transfer cask containing an MPC onto the SFP (event IDs 1 and 2). The frequency is assessed as very unlikely. These events are, nevertheless, retained for analysis because the consequences vary with the number of fuel assemblies present in the SFP at the time of the drop (which vary with the offload scenario) and because there is a large uncertainty in the rate of drops per transfer cask movement. These drops are evaluated for seismic events (event 9) only during periods when a transfer cask is being moved or held for decontamination over the SFP. Under these conditions, though unlikely, failure of the FHB steel super structure could conceivably cause the crane to fail dropping the transfer cask and having it impact the SFP.

Sum-2 is for the total non-seismic initiated frequency of any of the nine events resulting in any damage to fuel assemblies (IDs 1,2,3,5,6, and 7). The largest contributor to Sum-2 is event ID 5 at just more than half the frequency (aircraft crashes into the ISFSI). Event ID 2 is already considered in group Sum-1.

Sum-3 is for events 3 and 6 (also included in Sum-2) which are assessed as only leading to mechanical damage of the fuel assemblies within the storage overpacks. Mechanical damage events have the potential to release the radiation within the fuel rod gaps but radiation releases from the fuel itself are very limited posing no significant impact offsite. The frequency of these two events is also very low. Therefore, these two events are not considered further.

Sum-4 is for event 7 (also included in Sum-2) which is assessed as potentially leading to fuel overheating of the fuel assemblies within just one storage overpack. Event 7 has a low frequency of occurrence which was assessed conservatively but has the potential for a much greater consequence than for the mechanical damage events of Sum-3. However, the impacts of event 7 (tornados) are the same for all four offload scenarios being evaluated since loaded storage overpacks are already in place at the DCPP ISFSI. Therefore, event 7 is not considered further in assessing the comparative risks of the four offload scenarios. Its frequency is retained for use in comparing the absolute risks at the ISFSI event versus those at the SFP.

Sum-5 is for events 4 (seismic tipover of a storage overpack at ISFSI) and 5 (aircraft crashes). These events are assessed as potentially leading to fuel overheating of the fuel assemblies within multiple storage overpacks at the ISFSI. Again, the risks of these events are the same for all four offload scenarios being evaluated since loaded storage overpacks are already in place at the DCPP ISFSI. Therefore, events 4 and 5 are not considered further in assessing the comparative risks of the four offload scenarios. Their frequencies are retained for use in comparing the risks at the ISFSI event versus at the SFP.

Sum-6 is for seismic events 8 (seismic direct impacts causing SFP fuel uncovery) and 9 (seismic events causing drop of FHB crane and SFP fuel uncovery). These events are assessed as potentially leading to fuel overheating of the fuel assemblies within the SFP. The consequences of these events are dependent on the fuel assemblies present in the SFP at the time of the seismic event. Therefore, these events are evaluated considering the time-dependent fuel loading for each of the four offload scenarios. The contribution from event 9 is a minor addition to that of event 8.

In summary, the following events are considered in the assessment of comparative risks between offload scenarios (1, 2, 8 and 9). The additional events that are retained for comparison only of the absolute risks between the SFP and ISFSI are those potentially leading to fuel overheating (events 4, 5, and 7) (Sum-7).

Table 4-5. Summary of the Initiating Events Surviving the Screening in Appendix A and Their Estimated Frequencies per Year

ID	Sequence Class	Frequency/ Year	Consequence
1	Random drop of a transfer cask with MPC loaded with fuel assemblies for transport into the SFP causing a loss of coolant; summed over 80 lifts (total for both Units 2 and 2) and divided by an analysis period of 10 years	1.60E-09	Uncovery of fuel assemblies SFP at the time; potential for subsequent fuel overheating
2	Random drop of a transfer cask with empty MPC into SFP causing a loss of coolant. summed over 80 lifts (total for both Units 1 and 2) and divided by an analysis period of 10 years	1.60E-09	Uncovery of fuel assemblies in SFP at the time; potential for subsequent fuel overheating
3	Random drop of MPC loaded with fuel assemblies into a storage overpack at the CTF	2.20E-07	Mechanical damage to one MPC of fuel assemblies; no potential for fuel overheating
4	Seismic initiated tipover of storage overpack and resulting "damage" when initially anchored to ISFSI pad; conservatively bounded by assuming fuel assemblies within the storage overpacks overheat if "damage" occurs	2.80E-07	Fuel in five storage overpacks at ISFSI pad; fuel eventually assumed to overheat
5	Aircraft crashes into storage overpacks at ISFSI; conservatively bounded by assuming fuel assemblies within the storage overpacks impacted overheat if "damage" occurs	8.26E-07	Fuel in five storage overpacks at ISFSI pad fuel; fuel eventually assumed to overheat
6	Tornado missile strike on one storage overpack	1.00E-07	Mechanical damage to one MPC of fuel assemblies; no fuel overheating
7	Tornado winds block vents of one storage overpack at ISFSI	1.00E-07	Fuel in one storage overpack at ISFSI pad; assumed to eventually overheat.
8	Seismic events leading directly to fuel uncovery in the SFP	1.74E-5	Uncovery of fuel assemblies in SFP at the time of the seismic event; potential for subsequent fuel overheating
9	Seismic events causing FHB crane drop and resulting SFP fuel uncovery	8.54 E-9	Uncovery of fuel assemblies in SFP caused by seismic failure of FHB crane and drop of transfer cask being moved at time of event (total for both Units 1 and 2); 80 MPCs moved averaged over 10 years
10	Seismic events causing landslide into ISFSI storage overpacks	1E-7	Fuel in 20 storage overpacks assumed impacted by the postulated landslide sufficient to cause fuel overheating and release

Table 4-6. Sums of the Table 4-5 Initiating Event Frequencies per Year by Groups of Potential Consequences

ID	Sequence Class	Frequency/Year	Consequence
Sum-1	Random (i.e., non-seismic) frequency per year of fuel uncovery in SFP due to spent fuel storage and handling events; [IDs 1 and 2 (explicitly modeled)]	3.20E-09	Only a portion of fuel assemblies in the SFP at the time would potentially overheat. The number of fuel assemblies present varies with time and offload scenario.
Sum-2	Non-seismic failure frequency of mechanical fuel damage or fuel overheating per year for either the SFP or ISFSI locations, summed over all events; (IDs 1, 2, 3, 5, 6 and 7)	1.25E-06	Nearly all of this frequency is conservatively modeled to result in overheated fuel, see Sum-3 which is the frequency that only leads to mechanical damage
Sum-3	Frequency of mechanical fuel damage only events per year for the ISFSI location; (IDs 3 and 6)	3.20E-7	Mechanical damage to fuel rods resulting in gap release only
Sum-4	Frequency of fuel overheating involving the fuel assemblies in 1 MPC at the ISFSI; (ID 7)	1.00E-07	One MPC worth of fuel assemblies overheat at ISFSI
Sum-5	Frequency of fuel overheating at the ISFSI involving fuel assemblies in ~ five MPCs, summed over all events; (4 and 5)	1.11E-06	Most of the frequency involving fuel overheating is assessed as affecting multiple storage overpacks; five is assessed as a realistic number impacted
Sum-6	Seismic events leading directly or indirectly (due to crane drops) to fuel uncovery in the SFP; (IDs 8 and 9)	1.74E-5	The contribution of FHB crane drops (ID 9) contributes negligibly to the total seismic event caused SFP fuel uncovery frequency
Sum-7	Events at ISFSI that potentially lead to fuel overheating; (IDS 4, 5, 7, and 10)	1.31E-6	Most of the frequency involving fuel overheating at the ISFSI is assessed as affecting multiple storage overpacks; six MPCs are assessed as a realistic average number that are impacted by this collection of events

Table 4-7. Contributors to Risk at Three SNF Locations

Contributor Groups	Frequency of Fuel Uncovery while SNF is in Unit 2 SFP (per year)	Frequency of Fuel Assembly Mechanical Damage During Transport from FHB to ISFSI	Frequency of Fuel Loss of Cooling while in Storage at ISFSI (Per year)
Non-Seismic Initiators	3.2E-9 per year (SUM-6 or IDs 1 plus 2)	2.2E-7 (ID 3)	1.026E-6 (IDs 5,6, plus 7)
Seismic Initiators	1.74E-5 per year (SUM-6 or IDs 8 plus 9)	Screened as Negligible	4.80E-7 per year (IDs 4 plus 10)
Total Frequency at Each Location for All Initiators	1.74E-5 per year	2.2E-7 per year	1.51E-06 per year

Table 4-7 provides another way to look at the contributors to SNF risk; i.e., by location of the fuel assemblies. Three locations are considered; (1) stored in the SFP or being maneuvered within FHB, (2) transferred outside from the FHB to storage at the ISFSI, and (3) while in storage at the ISFSI. The numerical frequencies per year are presented by three groups of contributors: non-seismic initiators, seismic initiators including BDB seismic events, and their combination. These frequencies come from Tables 4-5 and 4-6 . Only those initiators in Table 4-1 that survive the screening exercises and are quantified as presented in Table 4-5 are considered here.

It is readily seen that on solely a severe accident frequency basis, the highest risk of a significant release is when SNF is within the FHB at the SFP with nearly all of the accident frequency coming from BDB seismic events. The accidents at this location are also of concern because an event involving uncovery of fuel within the SFP may lead to overheating and eventual release. The radionuclide release consequences may be large due to the large amount of SNF that an SFP typically contains and because the most recently offloaded batches of SNF to the SFP are at their highest levels of radioactivity. The USNRC also found the largest SNF risk contributor to be seismic events involving the SFP.

By contrast, the second location involving the transport of fuel assemblies from the FHB to the ISFSI has a much lower frequency of severe accident occurrence (only about 1% of the frequency of events involving the SFP). The frequency is for the transfer of all remaining fuel assemblies from both the Unit 1 and Unit 2 SFPs. This low frequency result comes from the design of the transport casks and the administrative controls in place at DCPP. Also contributing to the low frequency of severe accidents during transfers is the short duration of time of the transfer activities—only an average of four fuel loaded cask transfers per year amounting to about 12 hours duration per year total. While the cask transfers occur in discrete steps, the risk is normalized on a yearly basis in Table 4-7 for comparison to the other SNF locations. Years with more cask transfers would have higher frequencies while years with fewer transfers would have less. Another important consideration for this location is that the initiator quantified only leads to mechanical damage to the fuel within the one fuel loaded MPC being transferred. All fuel loaded MPCs are transferred only after at least 5 years of cooling time in the SFP. Often many of the fuel assemblies transferred in an MPC have cooled for much more than 5 years. Therefore the extent of a potential radionuclide release is much less than from an SFP overheating event. For these reasons, the risk contribution from this location is considered small and not evaluated further. Also, if all SNF is to eventually be removed from the SFP to the ISFSI, the transfer of fuel loaded MPCs to the ISFSI must occur for all four postulated offload scenarios and each offload scenario must remove the same number of fuel assemblies. This means the risk differences between the four defined offload scenarios should be small for this location. It is possible that risk differences could arise for different campaign strategies (all MPCs transferred consecutively, or in discrete batches separated by many months). This effect is not expected to be significant.

The third location is for the storage of fully loaded MPCs at the ISFSI. Each steel MPC is contained in a large, concrete-walled, storage overpack. As presented in Table 4-7, this location is vulnerable to non-seismic and seismic initiators. Here the non-seismic initiators are estimated to represent the highest risk. The assessment of these severe accident frequencies caused by external events judgmentally bounded the potential for the initiators to lead to overheating of the fuel assemblies within the storage overpacks. While likely more than one storage overpack may be affected (the total number of MPCs with overheating fuel would be a small fraction of the total present) it is estimated that no more than six MPCs loaded with fuel would be impacted as a result of any of these extremely low frequency events. Each of the initiators contributing to the potential for a severe accident at the ISFSI could in theory affect many storage packs at the same time. However, for each initiator in this group it is judged likely that only a small number would be affected and that many more would only be affected for a much less likely event than the one on which the occurrence frequency is based. As the number of loaded MPCs stored at the ISFSI increases, the risk of dry storage at the ISFSI is not expected to increase in proportion to the number of MPCs present. This seems counterintuitive but is an artifact of the nature of the severe accidents postulated and a bounding assessment of the number of MPCs affected for all times of dry storage at the ISFSI. For example, a light aircraft is expected to affect multiple storage overpacks at the ISFSI, but that number affected does not change if the number present increases from the current value of 87 to 137 planned after all fuel assemblies are removed from both SFPs.

Since each MPC holds at most 37 fuel assemblies (or 32 depending on the selected MPC design), the number estimated to potentially overheat would then be around 222 fuel assemblies. This corresponds to just a little more than one reactor core worth of fuel assemblies. Again, all fuel loaded MPCs are only transferred after at least 5 years of cooling time in the SFP. Often many of the fuel assemblies transferred in an MPC have cooled for much more than 5 years. Adding to the cooling time is the storage time at the ISFSI prior to the time of the severe accident occurring. The implication is that the consequences of such an event during storage at the ISFSI would likely also be much less than the release from a fuel uncovery event at the SFP.

Based on these insights from Table 4-7, this study focuses on the assessment of risks from storage of SNF at the SFP. The extreme accident frequencies at the SFP are higher than at the other two locations and the potential for larger radionuclide releases given a severe accident is also higher.

5 Description of Spent Fuel Pool Severe Accident

The issue important to SFP risk is the behavior of the spent fuel under accident conditions which result in the loss of coolant and uncovery of the fuel assemblies. Uncovery of the fuel assemblies can occur from an extended loss of SFP cooling without a leakage of coolant, via a leakage through the SFP liner. Safeguards available at DCPP to prevent these accident sequences from resulting in fuel uncovery, possible fuel overheating, and the consequential radioactivity release are described in Section 7.

On loss of water coolant sufficient to uncover at least a portion of the fuel assemblies, the temperature of the exposed fuel rises at a rate that depends on the amount of heat load in the fuel assembly and the configuration of the fuel. Table 5-1 lists some relevant temperatures for accident progressions following a loss of SFP coolant. In the absence of water coolant there is still air available to remove heat, but in many accident sequences it is an inadequate heat sink for preventing spent fuel overheating and thus the release of radionuclides.

Table 5-1. Event Temperatures for SFP Accident Progressions Involving a Loss of Coolant

Temperatures (°C)	Event Descriptions	References
>2300	All Noble Gases and Volatile Fission Products Released; (Iodine, Cesium, Bromine, Rubidium, Tellurium, Antimony, and Silver)	(Sehgal, 2012)
2227 - 2327	Fuel Collapses	(Sehgal, 2012)
>2000	Corium Melt Temperatures	(Sehgal, 2012)
1000-1500	Large Ruthenium Releases When in Air	(Sehgal, 2012)
>1227	Zircaloy-Oxygen Exothermic Heat Rate Is Large and May Become Self-Sustaining	(Sehgal, 2012)
1000-1200	Zircaloy-Steam Exothermic Reaction Begins	(Sehgal, 2012)
827-900	Zircaloy-Air Exothermic Reaction Begins	(EPRI, 2014)
727- 927	Zircaloy Cladding Balloons and Ruptures Occur	(Sehgal, 2012)
>800	Zircaloy Nitrogen Exothermic Reaction Begins (If No Air or Steam Present)	(Sehgal, 2012)

As fuel temperatures increase, there is an increased pressure in the fuel rods. At about 800ºC (in the range of 727 to 927°C) the pressures will cause the fuel cladding to exceed its yield stress leading to a phenomenon known as fuel ballooning. Fuel ballooning can occur at even lower temperatures because of thermal creep of the fuel rods (NASEM, 2016). Given that creep appears to be temperature dependent, fuel assemblies with higher heat generation would expect to be more vulnerable to fuel ballooning. Higher fuel assembly heat generation typically, but not always, is correlated with fuel assemblies which have higher burnups. In this assessment, while in the reactor prior to offloading to the SFP, the average fuel assembly is evaluated in a reactor core operating for 1200 consecutive days at a power level of 17.7 MW and with burnups up to approximately 50 GWD/MTU.

There are computational analyses suggesting that the peak cladding temperature can in fact be lower in high decay heat fuel assemblies than in neighboring lower decay heat assemblies, since there is initially higher steam production and therefore better cooling in the uncovered part of an SFP (NEA, 2015). There is considerable uncertainty about this potentially better steam cooling phenomenon. Should such not be the case, the result can be failure of the fuel cladding and a consequential release of gaseous and volatile fission products from the gap between the fuel rod and the fuel pellets.

The magnitude of the assembly source term depends on the power history and burnup of the fuel, that is, the gigawatt days of exposure per metric ton of uranium as well as the decay time since discharge. The fission products most likely to be released are (1) the noble gases xenon and krypton, (2) the volatile halogens, iodine and bromine, and (3) other volatile fission products including cesium, rubidium, tellurium, antimony and silver. At fuel temperatures in excess of 2300°C, nearly all of these fission products would be released. Studies to date indicate that for such SFP events, cesium-137 is expected to dominate the doses to the public and thus also the risk (USNRC, 2014; and EPRI, 2012).

It is important to observe that it is not necessary to have a zirconium-water reaction to have radiological releases. If the cladding is breached, a release of radioactive gases can occur that are already present in the gap between the fuel and cladding. But things do get much worse if the temperature rises. When the cladding temperatures reach the range of 900°C to 1200ºC, an exothermic metal reaction between the fuel rod cladding with steam or air, whichever is present, will occur. Should the cladding temperatures continue to rise, the reaction can become self-sustaining. The exothermic reaction heat rate may exceed the decay heat present at temperatures greater than 1230°C.

The oxidation can result in what is referred to as a zirconium cladding fire. A zirconium fire may propagate to other fuel assemblies in the pool and, in the limit, result in the release of most of the fission product inventory. Thus, the potential exists for a spent fuel pool accident to have a very large radiological release. However, as will be discussed later, there is evidence that this accident is of such low probability as a result of in-place safeguards that it is often screened out of consideration. But the phenomenological uncertainties are sufficiently high to warrant rigorous assessment.

Besides producing a great deal of heat, the zircaloy-water reactions produce large quantities of hydrogen. At sufficient hydrogen concentrations and if mixed with sufficient air, the hydrogen may burn or result in an explosion. Depending on the location, an explosion can damage the fuel handling building and the equipment it contains, thus hindering actions to mitigate the increasing fuel assembly temperatures and providing pathways outside the FHB for radionuclide releases.

In recent years, experimental evidence also indicates that there may be large releases of ruthenium radionuclides, if the fuel temperatures are between 1000°C and 1500°C and the fuel is exposed to air; i.e., if the steam is boiled off and there is a supply of air. While the release of ruthenium may be almost total if these temperatures are maintained and oxygen is depleted, the temperatures can only be maintained if the collapsed fuel has such sufficient decay heat to drive the molten corium-concrete interaction at the bottom of the SFP. At least one study using MAAP showed molten corium concrete reaction at the bottom of the SFP did not occur for a boil-off sequence without liner leakage (Fauske, 2019). The zircaloy cladding may also exothermally react with nitrogen at temperatures greater than 800°C, if the steam and the oxygen originally present in the air is already consumed (Sehgal, 2012).

Rigorous methods for assessing radiological releases and offsite consequences to support PRAs of nuclear facility accidents have been developed for nuclear reactors, but much less so for fuel assemblies in SFPs and dry cask storage facilities. For example, MELCOR (NEA, 2015, Appendix) is the basic code the USNRC uses to assess the consequences of nuclear power plant severe accidents. As noted earlier, the issue is the behavior of the spent fuel under conditions of the loss of SFP coolant water. There is considerable literature on the subject of natural air-cooling of fuel assemblies in an SFP under loss of all SFP coolant conditions (USNRC, 2014). The past research is problematical for DCPP, however, since most of the in-depth thermal-hydraulic analyses published to date have been performed only on BWR fuel assemblies when there is a substantial leakage through the liner at the bottom of the SFP at a rate that rapidly lowers coolant level within the SFP; e.g., equivalent to a 6-inch diameter hole. Depending on the arrangement of fuel assemblies within the SFP, the conclusions for BWRs seem to be that after 60 days within the spent fuel pool (since the most recent offload from the reactor) the fuel assemblies would be coolable in air by natural convection provided (USNRC, 2014) the following conditions exist:

- The hotter fuel assemblies are surrounded by cooler fuel assemblies in a 1x4 arrangement; i.e., each relatively high decay power fuel assembly surrounded by four lower decay heat power fuel assemblies not from the most recent reactor offload.
- The level of water within the spent fuel pool is low enough so that relatively cooler air can flow downward into the SFP by natural convection to the fuel rods by entering the coolant holes in the bottom of the fuel assemblies.
- There is sufficient ventilation or natural convection of outside air across the top of the spent fuel pool to the outside to carry away the decay heat being generated, thereby limiting the peak fuel cladding to temperatures below where Zr-air exothermic reactions do not occur.

For Diablo Canyon, a PWR with high-density racks, the 1x4 arrangement of fuel assemblies within the SFP is accomplished within 60 days of each fuel assembly unloaded from the reactor as required by the USNRC. See Figure 5-1 for a picture of the fuel assembly arrangement within the SFP. DCPP staff strive to accomplish the 1x4 arrangement sooner, time permitting—within the later stages of the refueling outage which is typically planned to last 33 days. During a refueling outage, the DCPP offloads the entire reactor core to the SFP. At the end of the outage, the fresh fuel and fuel not to be permanently stored in the SFP are returned to the reactor core. The 1x4 arrangement is only completed for the fuel which is not returned to the reactor core.

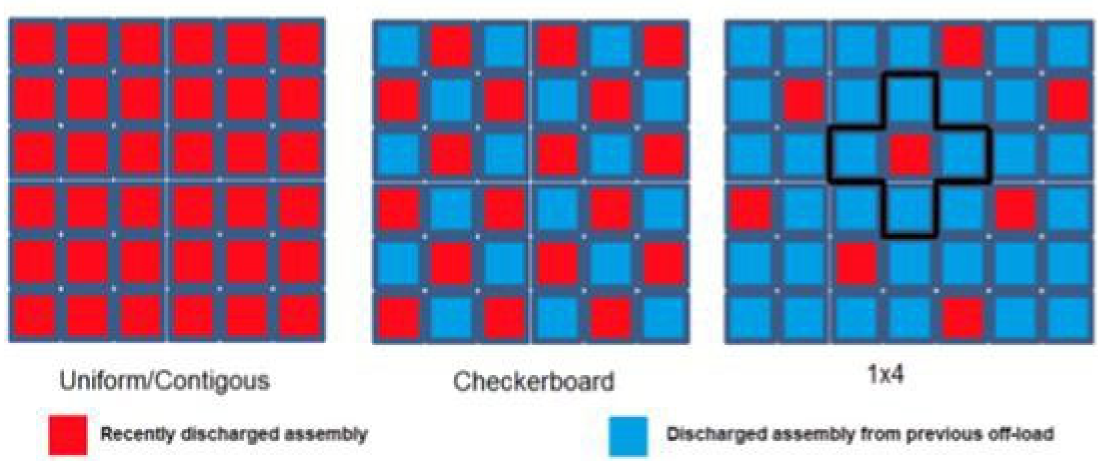


Figure 5-1. Alternative Fuel Assembly Arrangements Within an SFP

PWR fuel assemblies have higher power densities than BWR fuel assemblies and have closed storage cells within the SFP storage racks. For DCPP there is no initial steam or air crossflow between the fuel assemblies at higher elevations within the SFP as coolant level drops. Some have postulated that as the fuel assemblies and racks heat up, the racks would fail in such a way to enable such crossflows to occur before the coolant level drops below the bottom of the active fuel leading to more homogenous temperature rises (Fauske, 2019).

It is understood that MELCOR (Sandia, 2017, Sandia, 2017a) cannot fully represent such crossflow natural convection air flows and so definitive conclusions using MELCOR have not been reached for PWR fuel assemblies. Past analyses of spent fuel pool risks for PWRs have assumed fuel overheating conditional probabilities of 1.0, given such spent fuel pool loss of coolant sequences. Nevertheless, PWR plants are also subject to the requirements of B.5.b (USNRC, 2013) in that they must arrange the fuel in a 1x4 fuel assembly configuration shortly after reactor unload. Some PWRs have been exempted from this requirement due to other mitigating commitments, but DCPP does implement this 1x4 configuration. While some changes have been made in MELCOR to accommodate the analysis of spent fuel pools, it is still not an ideal tool because of the greater uncertainty in the results. To quote a recent National Academy of Sciences study (NASEM, 2016), "At present, there is no single modeling approach or software tool that can be used to model all aspects of loss-of-coolant accidents in spent fuel pools. Both integral models (MELCOR) and computational fluid dynamics (CFD) models have been used to examine spent fuel behavior in water-filled, partially drained, and fully drained pools. There are limitations to both types of models."

The lack of validation of MELCOR as a high confidence methodology for assessing releases and consequences of spent fuel accidents reinforces the need for rigorous assessment of the uncertainties in the results of any analysis of spent fuel pool risk.

SFP severe accident analyses performed with MAAP 5.0.1 (EPRI, 2013 and EPRI, 2014) have indicated that in the absence of air-cooling to limit fuel cladding temperatures, the release fractions of cesium could be very high, with MAAP results ranging from 25% to 96% released. However, the ability of MAAP 5.0.1 has not been validated for purposes of defining an acceptable water level in the SFP that precludes fuel damage to compute the times to accelerated exothermic reactions, nor to evaluate the benefits of fission product scrubbing by initiating sprays over the SFP. The cesium release fractions evaluated by MAAP as reported above have appeared in the literature but are not accepted as realistic for this study.

6 Release Characteristics and Selection of Risk Metrics

Radioactivity initially in the fuel assemblies stored within the SFP is in three locations; 1) the SFP water coolant, 2) the gap between the fuel cladding and the fuel pellets, and 3) the fuel matrix of the fuel pellets. The SFP cooling system and SFP skimming system limit the radionuclides which build up in the SFP coolant by removing contaminants. The radioactivity normally present in the SFP water coolant poses no risk to the public offsite. It must be monitored and managed to protect onsite workers, especially those working inside the FHB and around the SFP area.

If following the loss of coolant, the fuel cladding degrades or ruptures, a radioactivity release from the gap between the fuel cladding and the fuel pellets would occur. This allows the release of a portion of the more volatile radionuclides, such as noble gases xenon (xenon-135, half-life = 9.2 hours) and krypton (krypton-85 half-life = 10.8 years), iodine (iodine-131, half-life = 8.0 days), and cesium (cesium-137 half-life = 30.2 years; cesium-134 half-life = 2.1 years). These relatively volatile fission products have been released from the fuel matrix during normal reactor operation and accumulated in the gap still contained within the fuel cladding. The volatile radionuclides are still present in the gap after storage in the SFP. The gap amounts are a small percentage of the fuel inventory for non-volatile radionuclides and only a few percent of the volatile elements such as noble gases krypton-85 (2.9%), xenon-135 (0.16%); iodine-131 (1.9%); and cesium-134 and 137 (4.1%) (USNRC, 1995; USNRC, 2001a). Most of the fission products are still retained in the fuel matrix itself.

Once the fuel assemblies are removed from the SFP, they are loaded into MPCs for transfer to dry storage. The coolant is no longer a barrier to release. The MPCs, however, provide another pressurized metal barrier to release. The MPC is backfilled with helium to 30 to 40 psig at 70°F after evacuation of air and moisture. The MPC 32 container is designed to 200 psig for accidents (PG&E, 2018a, Section 4.2.3.3.2.2). From a mechanical damage event that breaks the MPC, the releases would be limited by the small amounts of noble gases and volatile fission products that have previously leaked outside the fuel cladding and were initially contained by the MPC. The mechanical damage event may also damage the fuel cladding which could then result in a gap release. These radioactivity amounts caused by gap releases are small and not expected to impact public health offsite.

At the ISFSI, the MPCs are stored within the storage overpacks. If air-cooling via the storage overpack vents is also reduced or eliminated, then there is a potential for fuel overheating. The rates of fuel overheating would be much slower than the hottest fuel assemblies stored in the SFP due to the much longer times since the fuel assemblies were offloaded from the reactor. Some fuel assemblies remaining in the SFP have been there for many years and would heat up at about the same rate as those in storage overpacks if SFP coolant were lost. For the BDB events considered, the releases from overheated fuel, as described for releases from reactor fuel assemblies are judged applicable for both releases from fuel assemblies stored at the SFP or the ISFSI.

At much higher temperatures, the releases of fission products from the fuel pellets occur. The radionuclides remaining in the fuel matrix can escape through the degraded fuel cladding and without coolant can be released to the containment, or if starting in the SFP to the FHB via spaces above the SFP and then potentially to the environment. The FHB is normally kept at negative pressure, but if the ventilation system stops due to a loss of AC power, the negative pressure difference with the environment would be lost. For severe accidents, when fuel temperatures exceed 2300°C, almost all of the noble gases (xenon and krypton) and volatile fission products are released (iodine, cesium, bromine, rubidium, tellurium, antimony, and silver) (Sehgal, 2012).

Semi-volatile fission products (molybdenum, rhodium, barium, palladium, and technetium) are characterized by high levels of release from the fuel pellets and are very sensitive to the oxidizing-reducing conditions (Sehgal, 2012).

The releases of low-volatility fission products (strontium, yttrium, niobium, ruthenium, lanthanum, cerium, and europium) from the fuel pellets are typically in the range of a few to 10% of the initial inventory (Sehgal, 2012). However, these releases can reach much higher levels for fuel with very high burnups. It is also known that the amounts released of ruthenium (Ru-103 half-life = 39.3 days, Ru-106 half-life = 373.6 days) can under some conditions be almost 100% in air even at temperatures in the range of 1000 to 1500°C.

For actinides such as uranium and neptunium, there can be as much as 10% released before the fuel matrix melts. Plutonium releases are always very low, less than 1%.

Cesium-137 (cesium-137 half-life = 30.17 years) is often chosen as a representative radionuclide for volatile fission products during severe accident releases in the long term. Large amounts of cesium-137 can also cause burns, acute radiation sickness, cancer, and even death. Iodine and ruthenium may remain largely in gaseous form, if there is substantial air ingress from outside the fuel handling building. Less volatile fission products tend to condense and form aerosols entrained with the steam and gas flow after release from the fuel matrix.

In Switzerland (ENSI, 2009), the amount of cesium-137 released is used to define a severe accident as resulting in a "large release"; i.e., greater than 2E+14 Bq. of cesium-137, where a Becquerel is a unit of radioactivity in which one nucleus decays by emitting radiation or disintegration per second. The same Swiss regulator defines "large, early release" by the amount of iodine-131 released; i.e., greater than 2E+15 Bq. of iodine-131 but only if the release occurs early in the accident sequence. "Early" is defined by the Swiss regulator as less than 10 hours following reactor core fuel damage.

As has been previously noted, iodine-131, which is also volatile and can cause thyroid cancers, has a half-life of just 8.0 days. Most fuel assemblies present in the SFP were offloaded from the reactor to the SFP years earlier, meaning that radioactive decay has already greatly reduced the total iodine-131 inventory in all but the most recently offloaded fuel assemblies. Only the most recently offloaded fuel assemblies (from the most recent refueling) would still have substantial quantities of iodine-131.

It is argued that the differences in risk posed by the four offload scenarios are only marginally affected by the short-lived radionuclides because of the generally long cooling periods prior to offloading to the ISFSI. Transport of fuel assemblies from the SFP to the ISFSI does not occur until at least 5 years after a fuel assembly has been removed from the reactor. By this time, the remaining activities of short-lived nuclides are not important. Further, there are no campaigns to the ISFSI within the first 110 days after reactor offload. Those fuel assemblies offloaded after 110 days are again from reactor offloads performed at least 5 years prior.

On the other hand, should an SFP uncovering event occur soon after the spent fuel is offloaded to the SFP from the reactor, the expected doses to the public would be affected by the short-lived radionuclides. For example, if a transport cask being loaded were to inadvertently fall within the SFP leading to a loss of SFP coolant and damage to fuel assemblies, a release could occur. The probabilities of such failures are very low and a transport cask would only be offloaded typically after more than 110 days of cooling following a refueling outage. The risks from seismic events are instead more governing. Seismic events can happen at any time and directly impact the SFP.

The offload scenarios vary in the timing for the transport of fuel assemblies decayed for more than 5 years to the ISFSI and hence in the number of older fuel assemblies present in the SFP at different times. The offload scenarios do not vary the length of time or number of fuel assemblies present in the SFP that still retain short-lived radionuclides because they are not transported until the short-lived radionuclides are decayed. Reactor risk studies have shown that iodine-131 (half-life of 8-days) typically governs risk when there has not been substantial decay time prior to release. After at least 110 days, the initial amount of iodine-131 has decayed to less than 0.01% of its initial value and would not contribute significantly to public health doses. Other nuclides with longer but still relatively short half-lives, such as ruthenium-103 (39.3 days) and ruthenium-106 (373.6 days) would still be present after 110 days but their release fractions are uncertain varying with fuel temperatures and the degree of air ingress. The release of cesium-134 (2 years) and -137 (30.2 years) are considered the significant dose contributors still present after 100 days.

Therefore, for measuring the risks of severe accidents involving the SFP or the ISFSI, iodine-131 is judged not to be a good representative radionuclide. The radionuclides of cesium (Cs-137 and Cs-134) have much longer half-lives and better represent the risk to the public and are used to differentiate the risks between offload scenarios. It is acknowledged that accidents initiated very early after offload (e.g., within the first 30 days), would also be affected by I-131 releases which have not yet decayed away. After only 30 days the inventory of I-131 has decreased by a factor of 1/16.

Figure 6-1 illustrates the decay of all forms of cesium within a representative fuel assembly as a function of time after reactor shutdown for two different fuel burnup amounts; see the red and blue curves. The burnup amounts refer to a measure of time spent in the reactor core while at-power and is a direct indicator of power produced by the fuel assembly while in the reactor. The burnup is measured in gigawatt days (1E+9 watt-days) per metric ton of uranium. This figure was presented by EPRI (EPRI, 2012), but is in turn developed from information from other sources (BSC, 2001; USDOE, 1992). The unit of cesium is presented as Tera-Bq per metric ton of heavy metal, times 1000. Tera-Bq is 1E+12 Bq. The initially steeper rate of cesium decay in the first 11 years after reactor offload is that from cesium-134 with a half-life of about 2 years. The slower rate of decay in Figure 6-1, after 11 years, is closer to the decay rate of cesium-137 which has a half-life of about 30 years.

Also shown in Figure 6-1 is the reduction of decay heat generation for the same representative fuel assembly versus time after reactor shutdown in years for the same two fuel burnup amounts. See the purple and green colored curves. The unit of decay heat on the right axis is kilowatts per metric ton of heavy metal. For the same burnup, it is easy to see that decay heat reduces more quickly during the first 5 years than does the total cesium decay. For the following 10 years decay heat is still reducing more quickly but the rates of reduction have narrowed considerably. After 15 years, cesium decay is reducing slightly more rapidly than is decay heat generation. Note that these curves are for a representative fuel assembly and not the SFP or ISFSI total heat load. This is because decay heat and isotopic inventory curves for each assembly is a function of the individual assembly power history, composition, and decay time. The total decay heat is the superposition of the decay heat for each fuel assembly remaining in the SFP at a given calendar time. After a period for cooling, fuel assemblies are then transferred to the ISFSI in different campaigns over a period of years.

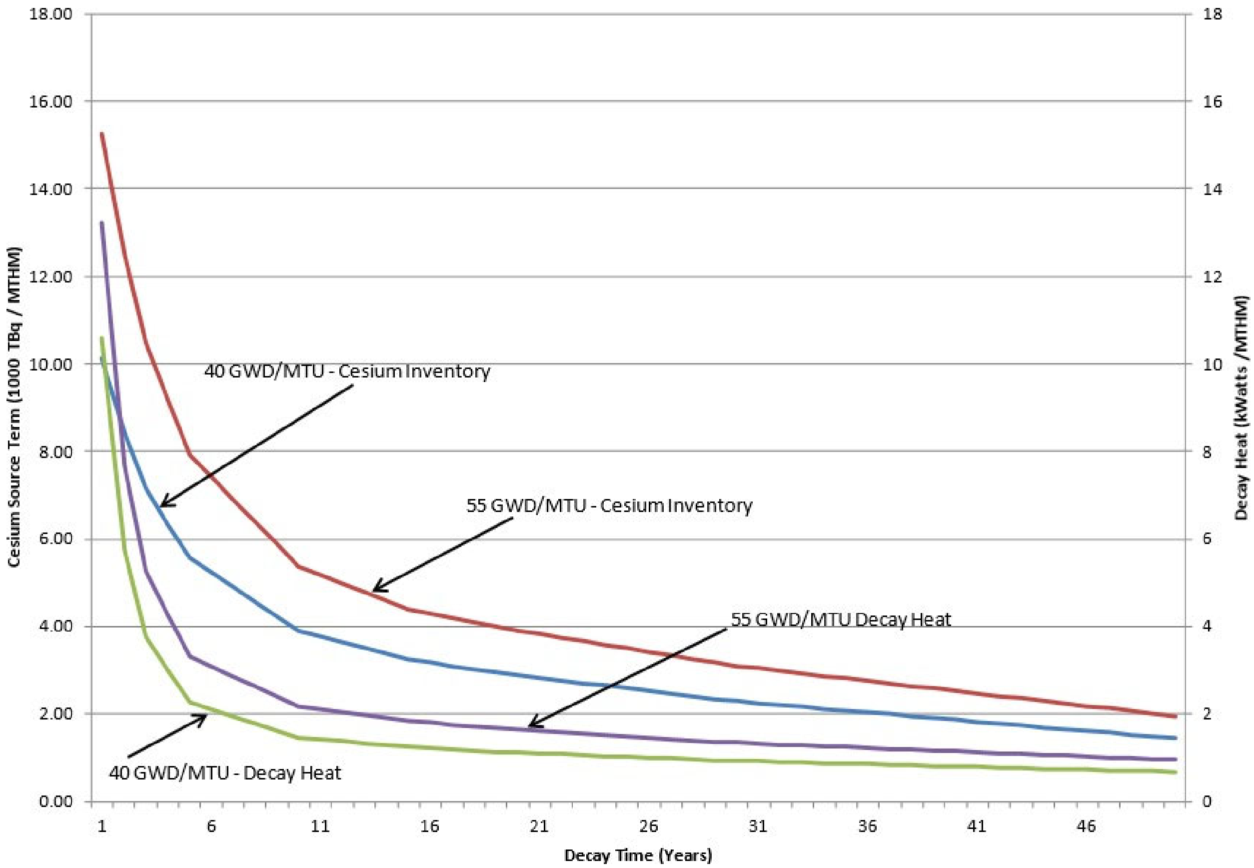


Figure 6-1. Time-Dependent Decay Heat and Cesium Activity for a Representative Fuel Assembly for Two Burnups

The potential impact on public health from a severe accident is a function of the total decay heat, power distribution among the fuel assemblies in the SFP, and the accompanying radioactivity available for release. For some sequences, the SFP decay heat levels can impact the allowable operator response and therefore the frequency of different sequences. For this assessment, the equivalent number of fuel assemblies remaining in the SFP at the time of the accident sequence, after adjusting for the radioactive decay of Cs-134 and Cs-137 for the times while in the SFP is considered as a first measure of the potential for radioactivity release. Sequences initiated after the final Unit 2 shutdown will typically have correspondingly less decay heat and again lower Cs-134 and Cs-137 activities.

Consideration of cesium decay while in the SFP was performed by tracking the fuel assemblies offloaded during each refueling outage and the dates the offloads occurred. An assumption was made that the fuel assemblies in the earliest refueling outages would have been transferred to the ISFSI first and in chronological order. Typically, these transferred fuel assemblies would have individual power levels of less than 600 watts each. This assumption of oldest assemblies being the first to be transferred to the ISFSI is an approximation as some fuel assemblies may have been transferred to the ISFSI with power levels between 600 watts and 898 watts per assembly. Also, some of the oldest fuel assemblies are still present in the Unit 2 SFP, so this assumed practice is not strictly followed. This assumption of oldest fuel assemblies being the first to be transferred eliminates the need to track each fuel assembly individually, thereby simplifying the analysis. The assumption underestimates the amount of cesium inventory transferred to the ISFSI, and therefore slightly overestimates the time-dependent amount of cesium activity remaining in the SFP. The assemblies in each reactor offload are decayed individually with time starting from an initial split of total cesium, 76% Cs-137 and 24% Cs-

134 (USNRC, 1987). By this approach, there are physically 848 fuel assemblies stored in the SFP at the start of this assessment period for each of the four offload scenarios, but the effective cesium inventory by equivalent fuel assembly count, after considering decay, is just 553 or 65% of the physical number of fuel assemblies.

For the fuel assemblies transferred to the ISFSI, each storage overpack is loaded with 32 fuel assemblies. These assemblies, as noted, have spent time cooling in the SFP. Using 5 years as the cooling time for the most recently transferred fuel assemblies, the remaining cesium inventory would be 72% of their initial cesium activity, corresponding to an effective fuel assembly count of 23. Of course, the storage overpacks which were first mounted at the ISFSI pads in 2009 were loaded with fuel assemblies first offloaded from the reactor in 1987. As of 2019, the initial level of cesium has decayed for 32 years, to 26% of the initial cesium activity, or an effective equivalent number of freshly offloaded from the reactor of just 8 instead of 32 fuel assemblies. More recently transferred fuel assemblies to the ISFSI would have effective fuel assembly counts between the range of 8 and 23.

There is another factor which potentially impacts public health from releases caused by a severe accident. In the SFP especially, the uncovery of fuel does not necessarily mean that the fuel present will all overheat sufficiently (possibly reach 2300°C where fuel collapse is expected) to release the fission products stored within its fuel matrix. Natural convection and passive radiation heat transport can limit the rise in fuel temperatures so that much of the fission product inventory may remain in the fuel matrix. This is especially true for the fuel assemblies that have been cooling the longest in the SFP. In essence, there is a conditional probability of each fuel assembly participating in release of radiation, given fuel uncovery, and this conditional probability varies with initial fuel assembly power level, cooling time, total SFP heat load, and arrangement of the fuel assemblies within the SFP. To determine these conditional probabilities requires a detailed computer model of the time- dependent response to a severe accident. Such a computer code would ideally consider the exothermic reactions of zirconium cladding with steam, air, and nitrogen as a function of time as the cladding comes in contact with each. Unfortunately, such a computer code does not exist that is sufficiently benchmarked to confidently account for all these variables. For this assessment, various alternative assumptions regarding the extent of fuel overheating given fuel uncovery are evaluated to assess the uncertainties in these conditional probabilities. A single assumption set is then selected to define our base case assessment of fuel overheating. See Section 8.

The fractions of initial radiological inventory released may in large part be determined by whether air-cooling by natural convection is successful for sequences involving fuel uncovery in the SFP. While procedures are available to align for FHB natural convection using outside air, the conditions present in these governing sequences are not favorable. Very large seismic events, which govern the frequency of SFP fuel uncovery, are judged to preclude operator actions, especially those requiring local actions at control stations outside the control room. While for most SFP severe accident sequences there is likely to be substantial time available to successfully respond, the time available will vary depending on the type of sequence; e.g., long term loss of all SFP cooling or leakage at various rates from the SFP. This assumption of no credit for operator actions above a seismic acceleration of 3.0 g SA is consistent with the at-power reactor risk studies performed to date for DCPP. It also recognizes that if a very large seismic event were to occur with the reactor at-power, there would be competition for resources to mitigate the impacts for both reactor units and both SFPs. In some sequences, radiation from a severe reactor accident may also preclude operator actions involving the SFPs. Less strong seismic events that may still fail the SFP liner and structure may also not allow sufficient time to align for SFP makeup or spray in time to avoid fuel uncovery, potentially leading to fuel overheating and release of cesium to the FHB.

In addition to numerous alternate sources of water to align for SFP makeup and cooling, at DCPP, there is a provision for aligning a seismically qualified water source with a fire hose to a portable crane with a long boom that could be used to spray the outside of the FHB without having to enter the FHB where radiation levels may be prohibitive. This provision would not prevent fuel uncovery since the spray is not necessarily directed into the SFP. Nevertheless, it could greatly reduce any cesium released from the fuel matrix to the FHB from being transported offsite. This provision has not been credited in the current assessment of releases offsite. Any future consideration of offsite public health impacts given cesium releases should consider this provision.

A major issue in all risk assessments is the choice of risk measures or indicators or indexes. For nuclear applications the risk measure is usually some variation of the probability of different levels of damage to human health, from radiation dose levels to injury from such exposures, to fatalities, and maybe even to property damage. Thus, there are usually multiple measures of risk. Typical risk measures are *frequency*, *probability*, or the *probability of frequency* of damage as a result of an undesired event. The convention considered most rigorous is the "probability of frequency" format. That is, the risk measure is the frequency of different levels of health effects or damage. The difficulty is that the frequencies are always accompanied with uncertainty, thus the entry of probability. Probability is the cornerstone of the science of uncertainty. It measures the credibility of an event based on the supporting evidence. The key is that probability must be anchored to supporting evidence. What this generally means in the rare event world is a probability distribution over the risk measure, usually a probability distribution of the frequency of a specific consequence. Such distributions are intended to tell much more of the real risk story than can be told with a single number.

So, what is the risk measure or metric in this risk assessment? Here we have compromised as the goal of the assessment precludes the need to conduct the so-called classic and rigorous probabilistic risk assessment. As has already been indicated, comparing risks enables major simplifications. Many simplifications have been identified such as using surrogates for consequences, radionuclide source terms, and of course risk measures. One major challenge is the time and condition dependence of the consequence of a threat such as a super earthquake or a heavy load drop over the spent fuel pool. The matter of time and condition dependence will be addressed later. For now, more on the simplifications made of the assessment.

Consequences are in terms of the amount of cesium activity released from the fuel matrix of the many fuel assemblies located at the SFP or ISFSI as result of a specific accident sequence. This simplification eliminates the need to assess the transport and health consequences of cesium and other radionuclides which may be released from the fuel assemblies to the offsite public. As explained earlier, this simplification is necessary because the available tools to perform a time- dependent assessment of cesium, and other radionuclides, have not been validated for the SFP and ISFSI.

Thus, sequence or scenario assessment becomes an elemental part of risk assessment. Typically, an accident sequence consists of an initiating event and a series of events that determine the trajectory of the accident sequence including additional failures and recovery actions while progressing through the sequence. Responses to the events must be considered as they are important in determining the consequence of a particular accident sequence. Such sequences are sometimes referred to as "what can go wrong" scenarios as noted by the triplet definition of risk presented earlier.

For this assessment, the potential initiating events were screened to limit the detailed sequence analysis to an assessment of seismic events. For seismic events, the impacts on the SFP systems and the ISFSI change as a function of the seismic acceleration range. Consistent with the DCPP reactor risk assessments performed, 16 ranges of acceleration were evaluated to cover the full span of seismic acceleration levels, including those acceleration levels much greater than the DCPP design basis.

Another aspect of defining accident sequences is an assessment of the initial conditions at the SFP and ISFSI which can determine the impacts posed by the different seismic initiating events and the potential cesium release consequences that may then result. This is particularly important for this assessment because the primary objective is to provide a basis for deciding between the risks of different offload scenarios. These alternate offload scenarios have differing numbers of fuel assemblies in the SFP and ISFSI with time. This, of course, also affects the heat loads at each SNF location with time and changes the amount of cesium activity present and hence potential releases. Seismic events occurring later in the storage period after longer cooling times typically have less potential to release large amounts of cesium due to cesium decay. Thus, the importance of time dependence in the assessment.

So effectively, the impacts of each of the seismic acceleration levels considered must be applied at each time interval while fuel is in the SFP or stored at the ISFSI. The results of the assessment which does exactly that for the SFP are described later in Section 8. The question is how to aggregate the results for each accident sequence while accounting for the time dependence.

One way to simplify the analysis is to make use of "pinch points" along the accident sequence. For there to be fuel overheating while in the SFP, there must be an uncovering of coolant normally covering the fuel. A pinch point can be defined for the intermediate sequence condition, "fuel uncovery". SFP fuel uncovery can occur as a result of sequences involving losses of coolant, extended loss of SFP cooling resulting in coolant boil-off, or both. The frequency of the intermediate sequence condition, fuel uncovery, is not a risk measure, but is a risk indicator and therefore a building block of a risk measure. Its drawback is that it does not yet distinguish the varying numbers of fuel assemblies present at the time of fuel uncovery and so cannot be considered a good measure of the risk for comparing alternate offload scenarios. An equivalent risk indicator for the ISFSI is the frequency of accident sequences potentially overheating fuel in a storage overpack mounted at the ISFSI pad.

To aggregate the results of all accident sequences requires the assessment of accident sequence consequences. As noted, an intermediate sequence condition is the equivalent number of fuel assemblies present in the SFP adjusted for cesium decay when a fuel uncovery event occurs. This represents one step closer to a meaningful risk measure but is not coupled to another important element of risk, namely the conditional probability of fuel overheating which is required for there to be a release of significant amounts of cesium from the fuel matrix. Nevertheless, this intermediate sequence condition is another way to consider the contributors to the frequency of potential cesium release as expressed by the equivalent number of fuel assemblies present in the SFP at the time of the fuel uncovery. Such investigations of contributors are considered in Section 8.3.

There is still the problem of accounting for time dependence. Consider the following. Perform a linear sum of the frequency of accident sequences resulting in fuel uncovery, multiplied by the duration of the time interval. Summed over all time intervals this yields the probability of SFP fuel uncovery while fuel assemblies are present in the SFP. These results change for different offload scenarios. This was accomplished for all time intervals while fuel assemblies are present in the SFP. At each time interval along the way the equivalent number of fuel assemblies present, adjusted for cesium decay, was also determined and multiplied by the probability of fuel uncovery for that same time interval. Summing these time interval products over all time intervals gives the probability of fuel uncovery weighted by an equivalent number of fuel assemblies present and adjusted for cesium decay. These integrated probabilities weighted by the equivalent number of fuel assemblies present are described in Section 8.3 and represent continuing progress towards an increasingly meaningful risk measure.

Such a risk measure still lacks completeness. Fuel assemblies that are uncovered do not necessarily overheat to the point of releasing cesium from the fuel matrix. The extent of fuel overheating is a complex function of numerous variables and some of these functions are not well known. They are also quite dependent on plant specific geometries, fuel assembly designs, and fuel assembly racks. Descriptions of what is known about the conditions which affect the potential for fuel overheating and the associated uncertainties are summarized in Section 8.4 and Appendix C.

It was decided that for offload scenario decision making, estimates of the extent of fuel overheating were required. Best estimate criteria for the extent of fuel overheating given fuel uncovery were developed for the two main sequence types leading to fuel uncovery, coolant leakage and non-coolant leakage events. The criteria applied depend on individual fuel assembly heat loads and the total SFP heat load at the time SFP fuel uncovery applies. The number of fuel assemblies determined to have overheated, adjusted for cesium decay, are then multiplied by the individual time interval fuel uncovery probabilities. These products are then summed up over all time intervals while fuel assemblies are in the SFP. Thus, the risk measure that is used in this study can be defined as *the probability of fuel uncovery weighted by time (and duration) and the equivalent number of fuel assemblies which overheat and release cesium, adjusted for cesium decay, given a severe accident including beyond design basis events.* The result is accountability for the time-dependent frequency of SFP fuel uncovery and the time dependent equivalent number of fuel assemblies that overheat, adjusted for cesium decay. This risk measure is reasonably complete as it varies with the offload scenario selected for evaluation. Indeed, this risk measure can be used to compare the different offload scenarios, the primary goal of this study.

7 DCPP Safeguards Against Spent Fuel Pool Severe Accidents

There is considerable confidence among the nuclear plant operators and the regulators that the SFP risks are extremely low and meet all the safety requirements of the Federal regulations. The primary reason for the confidence is the comprehensive practice of defense-in-depth against severe accidents which involves rigorous analyses to pinpoint vulnerabilities of the SFP to loss of coolant events. Spent fuel pools in general are among the most robust structures ever built (PG&E, 2017). The DCPP pools consist of 6-foot thick walls of reinforced concrete lined with stainless steel. The base of the pools is in bedrock. Shielding is provided by maintaining 23 feet of borated water over the top of the spent fuel. Extreme care is taken to assure the absence of penetrations to the pool which may be vulnerable to leaks or damage. For example, all piping associated with the flow and purification of the SFP water typically penetrates the spent fuel pool near the normal liquid level elevation. At DCPP, there are no piping connections below the 134-foot sea level, a level which is still 23 feet above the top of the fuel storage racks. See Table 7-1 for a listing of key SFP elevations.

Extensive safety design, safeguard features, procedures, and training make up the multiple layers of defense against severe accidents. The design bases consider severe earthquakes and massive impacts of extremely heavy loads. In addition, the systems for maintaining the water coolant height and temperature in the pools are designed for extremely high reliability and resilience making extensive use of the principles of redundancy and diversity. Besides the design features that are for the purpose of preventing and controlling catastrophic events, there are extensive procedures and systems to prevent or mitigate minor radiological consequences and operational risks.

DCPP has emergency procedures and systems in place to prevent a loss of cooling or loss of coolant inventory from the SFP long enough to lead to fuel uncovery should a beyond design basis earthquake or cask drop severely damage the SFP. The safeguards involved include multiple backup systems to the normally operating systems. These multiple backup systems require manual actions to align, but the large SFP water volume above the fuel and the associated heat capacity provide ample time to ensure the alignment of these safeguards when needed.

Part of the goal of this risk assessment is to investigate the impact of beyond design basis events. The key safeguard to a BDB event is the availability of makeup water to compensate for any loss of coolant event in the SFP. At DCPP the refueling water storage tank, condensate storage tank, and fire water storage tank are seismically qualified, essential long-term cooling water supplies. The raw water reservoir is not seismically qualified. Backups to the water storage and reservoirs include sitewide fire protection water systems but for makeup to the spent fuel pool, these are not seismically qualified.

Table 7-1. Key SFP Elevations

SFP Elevation	Description
140'	Top of SFP /operating floor around the SFP
139'	High level alarm (16" above normal)
138'6"	Highest elevation of SFP skimmer discharge piping penetration
137'8" to 136’8”	Normally maintained water level range
137'5"	Centerline of the return line penetration
137'4"	Low level alarm (4" below low end of normal range)
136’3”	0.5" hole in anti-syphon SFP cooling return line
134’8”	Centerline of the SFP cooling suction line strainer
134'0"	Centerline of SFP suction line piping penetration
133'	SFP water level designed not to fall below
123'11"	Low level 2 alarm; 10' above fuel rack
114'11"	Low level 3 alarm; 1' above fuel rack
115'0"	Elevation of transfer canal weir wall
113'11"	Elevation of storage rack top
111'11”	Elevation of top of fuel rods
99'	SFP bottom elevation (top of concrete)

The sources of water at the DCPP for coping with the many challenges of a severe threat that may result in a major loss of SFP cooling water are a good illustration of the concept of defense-in-depth. Among the multiple lines of defense are (1) various options for aligning different sources of water to the SFP, (2) the ability to add fire protection water to the SFP when normal makeup water is not available, (3) the ability to use a fire engine when normal makeup systems fail to perform their intended functions, (4) the ability to spray firewater over the SFP for cooling when coolant level cannot be maintained in the SFP or if the area around the SFP is not accessible, (5) methods to manage the firewater system to ensure sufficient capacity is maintained to carry out vital functions, and finally, (6) the use of FLEX equipment to assure such other needs as providing AC power for the equipment needed to perform the necessary extreme actions to assure SFP cooling. Adequate FLEX equipment is provided to support both reactors and both SFPs as might be required in a very large seismic event. After the reactors are permanently shut down, the FLEX equipment for the reactors is expected to be disposed of or sold. The FLEX equipment for the SFPs would not be required about 18 months after reactor shutdown, thus allowing 18 months for cooling in the SFP.

Control room alarms warn the operators of increases in SFP coolant temperatures or of low SFP water levels. In the event of a loss of SFP cooling, the annunciator response procedure (PG&E, 2019b) directs the suspension of any offload handling activities, especially for refueling outage offloads, if coolant temperature is greater than 125°F. If local checks do not immediately restore SFP cooling, the procedure directs the operators to transfer to the Abnormal Operating Procedure for SFP abnormalities (PG&E, 2019c).

A similar response would be directed in the event of a low SFP coolant level. In this case, however, there are three separate low-level alarms; the first one reached is at least a 4-inch drop from the then operating SFP water level. For such a drop in SFP coolant level, the SFP abnormalities Abnormal Operating Procedure (PG&E, 2019c) directs the operators to ensure that the fuel handling building ventilation system, which normally keeps the FHB ventilation at negative pressure with respect to the environment, switches from the normal operating mode to the emergency mode for iodine removal. If SFP water level drops to just 10 feet above the fuel assemblies (i.e., low level 2 alarm, about a 13-foot drop from the normal water level), then the operators are also directed to implement the DCPP emergency plan (PG&E, 2019d).

The SFP abnormalities Abnormal Operating Procedure (PG&E, 2019c) also directs that radiation be monitored in the SFP area, water be added to the SFP, and the SFP be checked for leakage paths. Otherwise, if SFP cooling pumps are available but SFP temperature still exceeds 140°F, then the steps in Appendix A for adding water to the SFP are to be implemented. There are numerous options listed in Appendix A. Borated water sources are preferred, but all but one of the borated water sources require pumps to move the water. The one exception is borated water from the refueling canal, which can flow by gravity if the fuel transfer canal is open and the cavity water level is above that of the SFP during a refueling outage when fuel assemblies are being moved in or out of the reactor building. If water is being added only to make up for boiling or evaporation, borated water need not be used. Regardless of whether borated water is used for SFP makeup, the operators are to check to ensure that boron concentration of the SFP coolant is at least 2000 parts per million.

Time-dependent calculations for the times for the SFP coolant to heat up to 200°F are periodically performed for the SFPs at DCPP (PG&E, 2019f). The time to 200°F is chosen as it is expected that for most cycle times the SFP would not reach full boiling temperatures but instead would evaporate at sufficient rate at 200°F to limit any further temperature increases while there is coolant in the SFP. Evaporation at these temperatures and rates would make for difficult access to the areas above the SFP to align for manual makeup or SFP spray from within the FHB. Figure 7-1 provides these conservatively evaluated times graphically for various assumptions. It is seen that early after reactor offload, the times available are much shorter but get progressively longer as cooling time within the SFP increases. Much of the total SFP decay heat comes from the most recently offloaded fuel from the reactor.

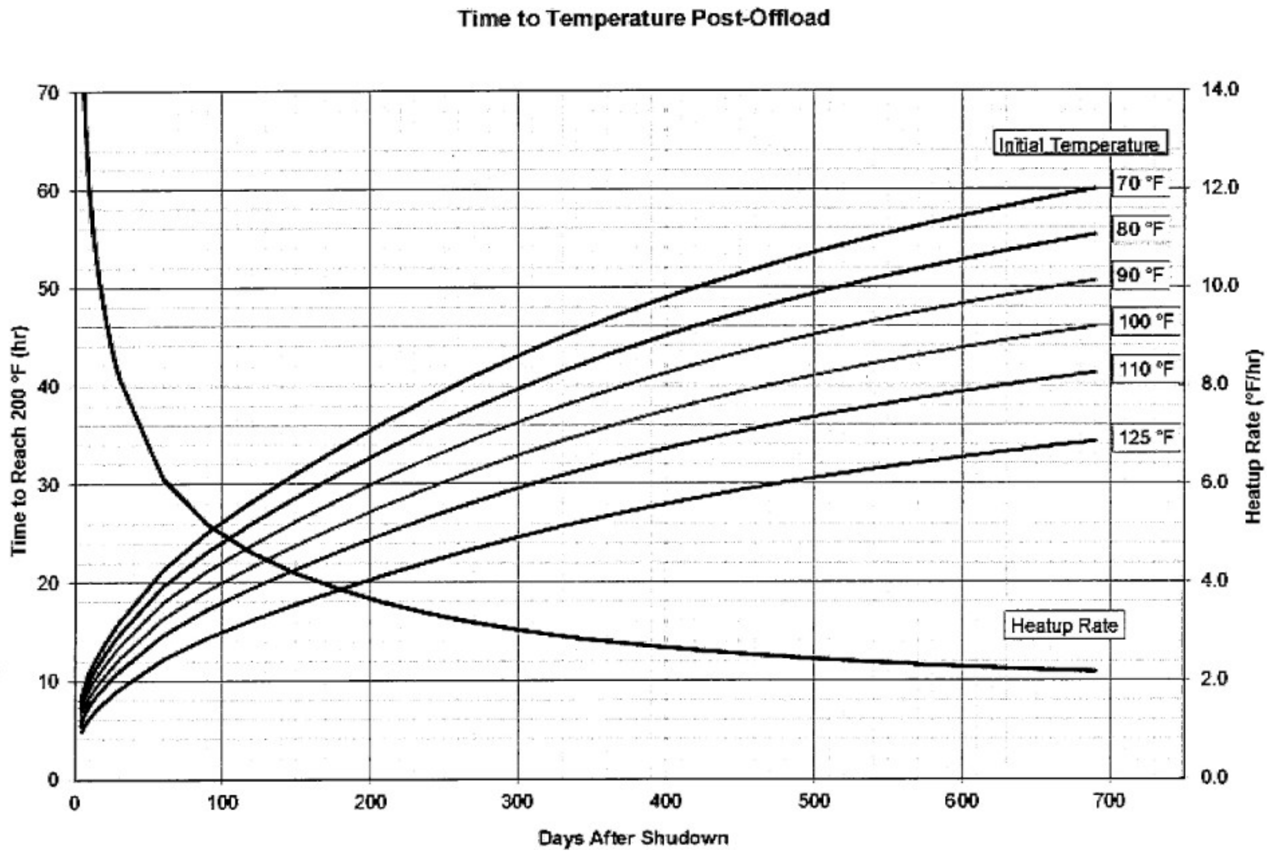


Figure 7-1. Estimated Hours for SFP Heatup to 200°F from a Given Initial SFP Coolant Temperature after a Full Core Discharge

If at any time an extreme damage event occurs (e.g., when a leak through the liner of the SFP is detected and not easily isolated), then SFP leakage control strategies are considered to patch the leak (PG&E, 2019e). Such patches have been made to other plant's SFP liners, but typically these are for holes located relatively high in the SFP. This procedure lists four options to patching the detected liner leak location. Area radiation monitoring is to be conducted and these strategies are not to be implemented if SFP coolant level drops to just 8 feet above the fuel racks, below which radiation levels may be dangerously high. The procedure with these leakage control strategies was written assuming a limiting 500 gallons per minute SFP leak, which would allow 5 to 6 hours before level drops to 8 feet above the fuel racks.

The Unit 2 SFP bottom liner is located at the 99-foot elevation. The cask recess area extends down another 4 feet 6 inches. The floor of the FHB outside the SFP is at 100 feet. The ground level elevation on the east side is at 115 feet. This is the side that the low-profile transporter rolls the loaded fuel casks in a transfer cask outside. While the base of the FHB has no other floors directly below the SFP, there are lower levels within the same building on the west side, down to the 85-foot ground elevation. Building levels on that side extend down to 54 feet where the pipe tunnels are located. It is expected that if SFP leakage occurs to the surrounding SFP area, that SFP coolant may be lost to these lower building levels. Loss of SFP coolant to the 100-foot level adjacent to the SFP may complicate the actions to manually align for SFP makeup and spray.

Section 6, Appendix B, of the SFP abnormalities Abnormal Operating Procedure to provide enhanced ventilation (PG&E, 2019c) may be implemented for SFP cooling by evaporation to achieve safe working conditions in the SFP area, or if the SFP water level drops below the top of the active fuel. This added ventilation is achieved by opening FHB doors to provide air cooling. There are several doors available for establishing this ventilation. If the FHB ventilation system is not available, these may include the fuel receiving rollup door at the 115-foot level, either of two rollup doors (525 or 526) in the FHB, and the FHB north or south stairwell doors (530 or 530-2) at the 140-foot level to enhance flow across the SFP coolant surface. Portable fans may also be installed in the doorways to provide more flow. If radiation levels are too high for FHB access at the 140-foot level to open doors, then a destructive path through the FHB roof may be opened to provide increased natural convection flow.

FLEX guidelines (PG&E, 2019) are entered from the Abnormal Operating Procedure “SFP abnormalities” (PG&E, 2019c) when there is no AC power to supply pumps and the ultimate heat sink is also lost. Such impacts to DCPP may occur from a strong seismic event. If a large opening is also created in the SFP liner by a seismic event, the time available for implementing the steps in this guideline may be limited. This procedure makes use of the FLEX suction header water supply as another source of water for makeup or spray to the SFP. The preferred method of makeup is using hoses to the 140-foot level of the FHB to make use of a pre-staged connection in the stairwell. If the 140-foot level is not accessible, then water can also be added at the 100-foot level from connections inside the FHB, but from a location just outside the SFP area near the SFP cooling pumps. Piping connections for these alignments have been pre-planned.

Separate from Section 6, Appendix B, of the Abnormal Operating Procedure “SFP abnormalities” (PG&E, 2019c), if radiation levels at the 140-foot level are greater than 1 rad per hour, then Section 6 of the FLEX guideline may also be implemented to provide SFP cooling by evaporation. Natural convection cooling using outside air is established through the FHB by opening doors at both the 115-foot and 140-foot elevations. An exception to the guidance to implement this natural convection alignment is if the accident occurs within 30 days of having just offloaded fuel and the SFP coolant level has already dropped more than 23 feet to just below the top of the fuel. In this case, the FLEX guidelines direct that the FHB doors are to be maintained closed limiting air access to prevent a zirconium-air fire. If the accident occurs more than 30 days after fuel offload, then the doors would be opened in accordance with these guidelines.

8 Analysis of Moving SNF from the SFP to the ISFSI

This assessment addresses the following six questions, the direct responses to which are in Section 9, Conclusions.

The purpose of this chapter is to summarize the basis for the responses:

1. *Which offload scenario is the least vulnerable to a beyond design basis radiological event, that is, which scenario of the four considered is most likely to assure public safety? How do the four offload scenarios rank?*
2. *How does the difference in risk between the SFP and the ISFSI impact the safest operating strategy?*
3. *What additional technology is needed to perform a full scope probabilistic risk assessment of the SFP and the ISFSI with the same confidence as those typical of contemporary nuclear power plant PRAs, particularly with regard to the quantification of the uncertainties?*
4. *How does the current analysis compare the spent fuel risk with the DCPP risk?*
5. *What are the most important variables affecting SFP risk?*
6. *How do the uncertainties in the assessment impact the conclusions?*

8.1 Assessment of the Four Offload Scenarios

The four alternative offload scenarios to dry storage are briefly described in Sections 1, 2 and 3. Figure 8-1 displays the scenarios graphically by illustrating the time-dependent number of fuel assemblies in the Unit 2 SFP for each offload scenario. Roughly 80 fuel assemblies are permanently transferred from the reactor core to the SFP during each refueling. This is the assumed amount offloaded each refueling for all four offload scenarios. This illustration is an idealization for simplicity. During the reactor offloads in each refueling outage, DCPP actually offloads the entire core of fuel assemblies to the SFP for a 2-week period, and then reloads the reactor core. Only the changes in permanently offloaded fuel assemblies is shown in Figure 8-1. The temporary peaks for the full offload are not shown. These temporary periods of full core offloads are accounted for in the risk results presented throughout this section.

It is seen that each of the alternative offload scenarios is the same until the beginning of Unit 2 refueling outage 2R22, which is scheduled for May 2, 2021. The analysis for this assessment compares the risk results for times after May 2, 2021, and until all fuel assemblies are removed from the Unit 2 SFP.

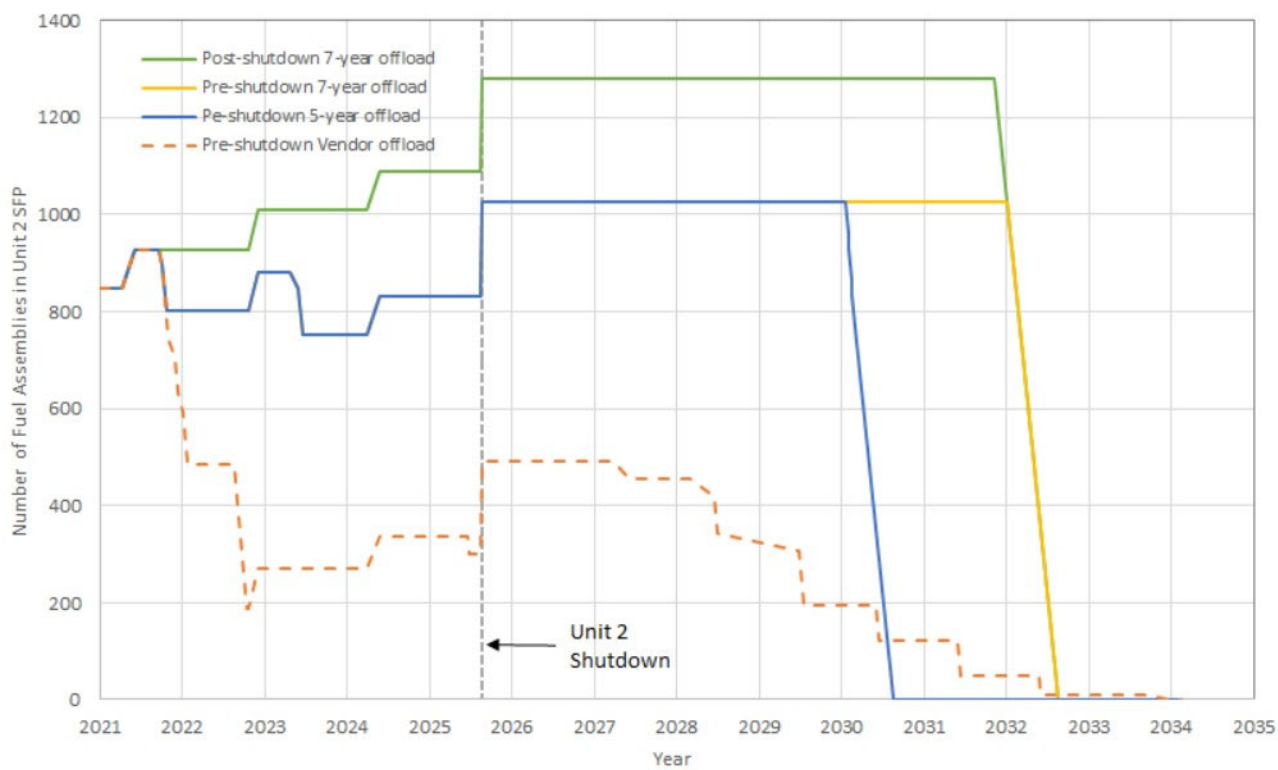


Figure 8-1. Time-Dependent Number of Fuel Assemblies in the Unit 2 SFP Versus Calendar Year for Four Offload Scenarios

Table 8-1 presents some insights for the four offload scenarios regarding the number of fuel assemblies present. There are to be three reactor refueling periods starting May 2, 2021, with 2R22 and including 2R23 and 2R24, to be followed by the final Unit 2 shutdown in late August 2025. These reactor fuel assembly offloads are the same for all four offload scenarios to dry storage. Therefore, the number of fuel assemblies initially present in the Unit 2 SFP, the number still to be transferred for the reactor, and the total number to be offloaded to dry storage are also the same for all four offload scenarios.

The four offload scenarios have varying amounts of fuel offloaded to the ISFSI prior to the EOL so that the number of fuel assemblies present at the time of EOL and the peak number of permanently offloaded fuel assemblies varies with the offload scenario. Three of the four offload scenarios have their peak number of fuel assemblies at the EOL. For the vendor option offload scenario, the maximum number of permanently offloaded fuel assemblies is after reactor offload in refueling outage 2R22. This is the only offload scenario to transfer more fuel assemblies to the ISFSI prior to EOL than are offloaded from the reactor between 2R22 and EOL.

The last column of Table 8-1 refers to the equivalent number of fuel assemblies after adjusting for cesium decay. For this column, only the post-shutdown 7-year offload still has a maximum at the EOL. The other three offload scenarios peak at 2R22, before any fuel assemblies are transferred to the ISFSI. For all four offload scenarios, the equivalent number of fuel assemblies is much less than the number physically present. The differences between the last two columns show the reduction attributed to cesium decay. This should be considered when trying to compare the relative potential for release from the SFP versus that from the reactor core. No more than 4.5 equivalent reactor cores worth of cesium are present in the SFP for all four offload scenarios. The movement of fuel assemblies for Unit 1 is expected to follow the same four offload scenarios, although progressing and ending a year earlier.

Table 8-1. Impact of Different Offload Scenarios on Unit 2 SFP Inventory

Offoading Scenarios (Years refer to SFP empty time after Unit 2 EOL)	# of Reactor Fuel offloadings beginning 5/2/2021	# of Fuel Assembles in SFP on 1/2021	# of Fuel Assemblies offloaded from Reactor after 5/2/2021	# of Fuel assemblies to be Transferred to Dry Storage after 5/2/2021	# of Fuel Assemblies in SFP after Unit 2 final offload (EOL, 8/26/2025) (Equivalent Cores)	Maximum Fuel Assemblies in SFP Permanently offloaded (Equivalent Cores. Peak date)	Maximum Equivalent Fuel Assemblies-adjusted for cesium decay
Post-shutdown 7-year offload	4	848	433	1281	1281 (6.64)	1281 (6.64, EOL)	870 (4.5, EOL)
Pre-shutdown 7-year offload	4	848	433	1281	1025 (5.31)	1025 (5.31, EOL)	746 (3.9, 2R22)
Pre-shutdown 5-year offload	4	848	433	1281	1025 (5.31)	1025 (5.31, EOL)	746 (3.9, 2R22)
option offload	4	848	433	1281	492 (2.55)	928 (4.81, 2R22)	746 (3.9, 2R22)

Table 8-2 presents some additional insights for the four offload scenarios, this time in terms of the multipurpose cask movements to dry storage for Unit 1 and Unit 2. The total number of fully loaded MPCs to be transferred is the same for three offload scenarios, but the number transferred is smaller for the pre-shutdown vendor option. This is due to the different capacities of the MPCs assumed to be used. The vendor option MPC holds 37 fuel assemblies while the MPC used for the other three options holds just 32. The number of campaigns to transfer fully loaded MPCs to dry storage differs both in total number and in terms of whether they occur before or after Unit 2 is permanently shut down relative to the date of EOL. It is noted that a future decision on the MPC design to be used for future campaigns has not yet been made. The greater capacity of the MPC37 design is attractive because it would free up ISFSI locations to accommodate other types of waste. It is likely that whichever MPC design is chosen for future campaigns, the selected MPC type would also be used for the selected offload scenario concept. The numbers in Table 8-2 are for offloading the Unit 2 SFP. The total number of MPCs transferred is doubled in the last column to reflect the total for both Unit 1 and Unit 2.

The vendor option is very different from the other three offload scenarios in that its largest offload campaigns take place before EOL, whereas the largest campaigns for the other three offload scenarios occur in the final campaign, ending with no fuel assemblies left in the Unit 2 SFP. The pre-shutdown vendor option offload scenario has a total of nine separate campaigns−three times any of the other three offload scenarios. The last four of the nine vendor offload scenario campaigns involve one or at most two MPCs to be transferred to dry storage. While the pre-shutdown vendor option offload scenario moves a greater number of fuel assemblies out of the Unit 2 SFP at the earliest possible dates, its total duration of having fuel assemblies remain in the SFP is longest. In particular, the vendor option offload scenario is 12.7 years, as measured from May 5, 2021, versus 9.3 years for the pre-shutdown 5-year offload and 11.3 years for the other two offload scenarios. The result is an increase in the time that some fuel assemblies are still in the SFP. This increases the chance that a large seismic event occurs with fuel assemblies still in the SFP for the pre-shutdown vendor option offload scenario.

The exposure durations for each offload scenario are illustrated in Figure 8-2. There are 193 fuel assemblies that make a full reactor core of fuel for the DCPP. Therefore, the amounts of fuel in

Table 8-2. High Level Characterization of Offload Scenarios for Unit 2 in Terms of MPC Movements

Offoading Scenarios (Years refer to SFP empty time after Unit 2 EOL)	Planned Duration with SNF in SFP after 5/2/2021 (years)	Date all fuel assemblies are removed from SFP	Number of MPC Campaigns to Dry Storage, per Unit Before/After EOL	MPCs Transferred for each Campaign Before/After EOL	# of Unit 1&2 MPCs to be Transferred to dry storage
Post-shutdown 7-year offload	11.33	September, 2032	0/1	0/40	80 (32 FAs each)
Pre-shutdown 7-year offload	11.33	September, 2032	2/1	4,4/32	80 (32 FAs each)
Pre-shutdown 5-year offload	9.33	September, 2030	2/1	4,4/32	80 (32 FAs each)
Pre-shutdown vendor option offload	12.69	January, 2034	2/5	12,8,1/1,7,2,2,1,1	70 (37 FAs each)

Figure 8-2 are expressed in ranges of equivalent reactor cores. It is seen that the pre-shutdown vendor offload scenario (shown with orange bars) has the advantage of spending the most time in the Unit 2 SFP with less than three reactor core equivalents of fuel assemblies. By contrast, the other three offload scenarios spend very little time with this low-end range of fuel assemblies in the SFP.

The post-shutdown 7-year offload scenario (green bars) is the only option among the four which spends time with more than six equivalent cores of fuel in the SFP, and the duration compared to the other offload scenarios is long, more than 6 years of exposure after May 2, 2021. For this same offload scenario, almost 3 years is also spent with five or six cores of fuel assemblies in the SFP. The other two offload scenarios (pre-shutdown 7- year (yellow) and pre-shutdown 5-year (blue)) spend the most time with three, four, five, or six core equivalents of fuel assemblies in the SFP. Only the 7-year offload scenario option has more than 7 equivalent cores of fuel assemblies in the SFP, and only for a relatively short time. From Figure 8-2 it is concluded that if a large seismic event occurs while there are fuel assemblies in the SFP, the pre-shutdown vendor option offload scenario has the advantage by limiting the amount of fuel assemblies that can potentially be impacted.

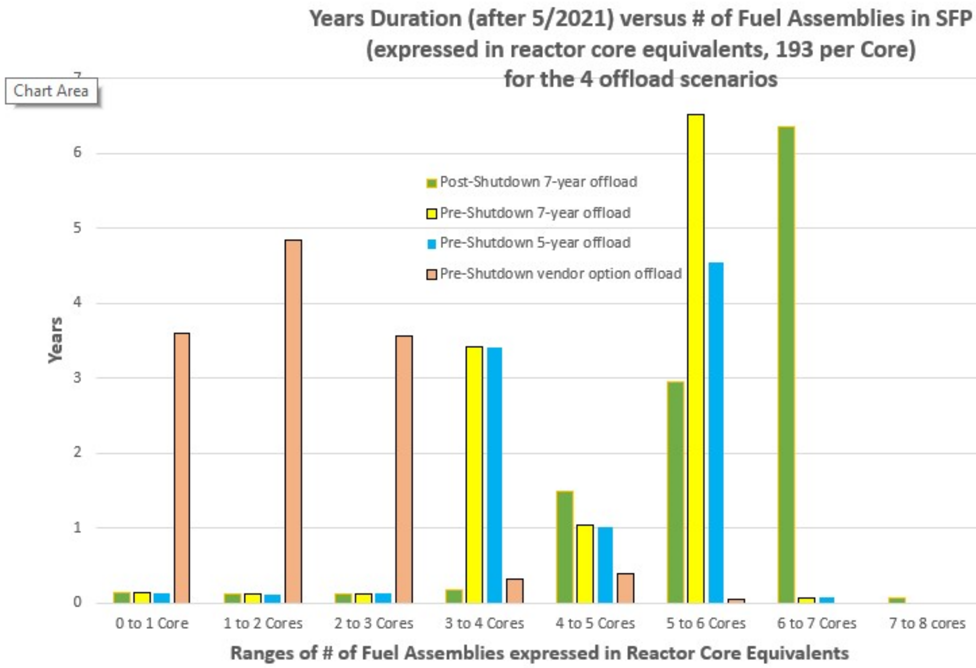


Figure 8-2. Years Duration versus Number of Fuel Assemblies in the SFP Expressed in Reactor Core Equivalents for Four Offload Scenarios

The duration of exposure for all fuel assemblies present in the SFP does not tell the full story. Fuel assemblies with longer cooling times in the SFP have lower decay heat power and lower fission product inventory. It is expected that lower decay heat fuel assemblies are also less vulnerable to overheating if there is a fuel uncovery event. For these reasons, the fuel assemblies of interest to fuel overheating and cesium release may instead only be those within the SFP that are offloaded in the most recent years. These most recently offloaded fuel assemblies then pose the greatest potential for overheating and therefore offsite radiological releases.

Figure 8-3 shows how the durations change when only fuel assemblies cooled for less than 3 years in the SFP are counted. Each of the fuel assemblies excluded from this figure has been cooling for more than 3 years. Figure 8-3 shows that durations for zero to one core, and for one to two cores are the same for the four offload scenarios. All four offload scenarios indicate zero durations for more than two equivalent reactor cores of fuel assemblies when the fuel assemblies counted are only those cooled for less than 3 years. This observation indicates that if fuel assemblies that have been cooled for more than 3 years in the SFP can be shown to not overheat and release cesium, then there is essentially no difference in the risk from fuel overheating between the four offload scenarios.

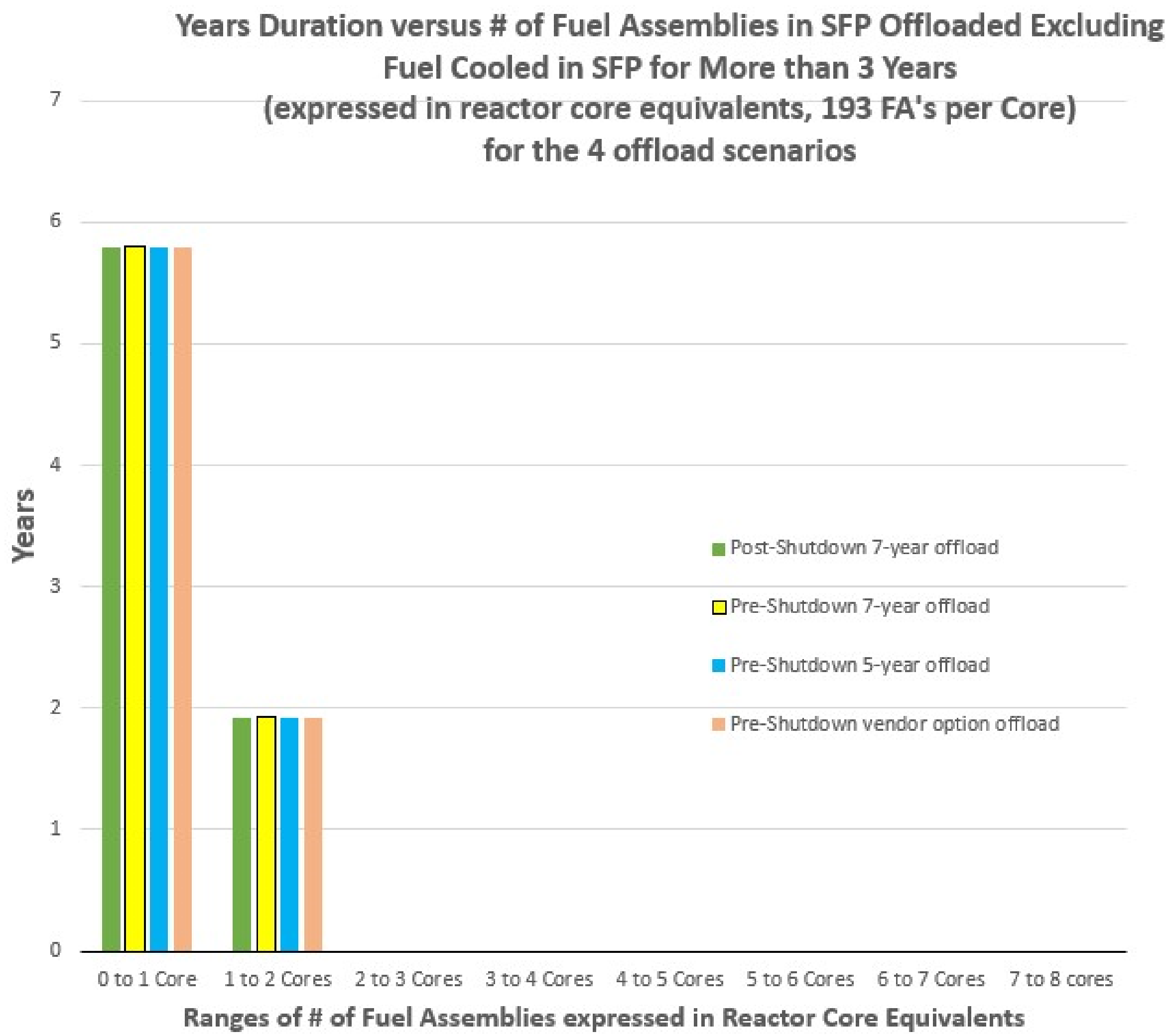


Figure 8-3. Years Duration in the SFP for Fuel Assemblies Restricted to those Offloaded Only Since 5/2016 for Four Offload Scenarios

8.2 Comparisons of Frequencies of Initiating Event Contributors

The seismic frequency of fuel uncovery while assemblies are in the SFP includes the contribution of all ranges of seismic magnitude (below and above the seismic design basis levels) and accounting for the probabilities of seismic failure modes for key DCPP systems, structure, and components.

Components considered for seismic failure modes include:

1. AC power from offsite
2. Backup AC power from emergency diesel generators
3. Auxiliary building structure
4. FHB steel super structure (supporting the crane)
5. SFP liner and concrete structure
6. Onsite 4kv and 480v distribution network
7. 125v DC power

8. FHB ventilation equipment
9. SFP cooling component cooling water

Human error probabilities for selected mitigative actions were increased for higher seismic accelerations consistent with the models for reactor at-power risk (PG&E, 2019a). There are three contributors to the total human error probabilities when increasing the human error probabilities for stronger seismic events; 1) the cognitive decision to perform the action, 2) travel to the required plant location, and 3) the time required to execute the intended action. The error probability for the decision to perform the needed action is not expected to increase with seismic acceleration, but for actions to recover from a loss of SFP cooling, travel to the SFP is necessary to investigate the need to restore cooling.

The execution time might be longer after a seismic event, but most of the actions credited for SFP cooling recovery are fairly simple and are routinely performed for normal tasks, such as opening a valve, opening doors, or closing a breaker. The involved component must also be functional for the operator action to be credited. Unlike the execution time, the travel time from the operator's starting point to the intended location will very likely be impacted by a seismic event. Potential obstructions or stuck doors may require the operator to take a longer path to reach the location of the action. Increased operator stress and lack of lighting are likely to impact the travel time for beyond design basis seismic events.

The model for high seismic acceleration impacts on human error probabilities for SFP recovery actions adopted for this study is the same as that used in the DCPP reactor risk study. The total human error probability which considers cognitive time, travel time, and execution time is modified by a designated multiplier depending on the seismic acceleration level. The human error probability for accelerations less than 1.75 g SA are taken to be the same as for non-seismic events. Between 1.75 g SA and 3.0 g SA the total human error probabilities are multiplied by a factor of 5. For seismic accelerations greater than 3.0 g SA, (just less than twice the design basis seismic acceleration) no credit is given for the human actions to recover from a loss of SFP cooling or a loss of coolant.

Figure 8-4 presents the seismic event tree structure that considers the seismic failure modes and recovery actions to determine the sequences leading to fuel uncovery. Although 16 different seismic acceleration ranges were quantified separately, the sequence frequencies presented in Figure 8-4 are for the sum over all acceleration ranges. The total frequency of SFP fuel uncovery is 1.74E-5 per year for periods while one or both of the DCPP reactors are operating and slightly less (1.69E- 5 per year) when both reactors are not operating but some fuel assemblies are still in the Unit 2 SFP. Significant severe accident releases from a concurrent accident at an operating reactor are assumed to impact the potential for operator mitigation of seismic impacts on the SFP. This point is described further in the presentation of Figure 8-8 below which displays the impact on SFP fuel uncovery frequency versus seismic acceleration level.

Initiator	FHB/ AUXB Intact 2.3g HCLPF	SFP Intact, No Moderate Breaks (1) 3.54g HCLPF	SFP Intact, No Small Breaks 3.54g HCLPF	AC Power Available <2.06g HCLPF	SFPC SFP Cooling HCLPF=2.50g	NEM Normal and Emerg.. Makeup	B5b FLEX Makeup	HYB No H2 Burn	FHB Ventilation HCLPF = 1.49g	Sequence ID	Consequence	Sum of Seq. Frequency SEIS1-SEIS16 (per calendar
Earthquake	Yes	Yes	Yes	Yes	Yes	NN	NN	NA	NN	1	No release	2.02E-02
PER CAL.YEAR	(1-SAB)	(1-SFP*0.5)	(1-SML)	(1-S48)	(1-SFPC) No	Yes	NN	NA	NN	2	No release	1.11E-05
					SFPC	(1-NEM1) No	Yes	NA	NN	3	No release	5.47E-08
						NEM1	(1-B5b1) No Makeup	NA	Yes	4	Filtered SFP release	1.12E-08
							B5b1		(1-VEN) No	5	SFP full release	5.49E-07
				No AC	NA	NA	Yes	NA	VEN NA	6	No release	1.98E-05
				S48			B5b2 No Makeup	NA	NA	7	SFP full release	1.09E-05
			Smaller Leak Drains to bottom of pool	Yes	NA	Yes	B5b2= .002 NN	NA	NN	8	No release	1.11E-06
			SML=SFP *0.5	(1-S48)		(1-NEM2) No makeup	Yes	NA	NN	9	No release	1.50E-08
						NEM2	(1-B5b1) No Makeup	No BURN	Yes	10	Filtered SFP release	1.45E-09
							B5b1	(1-HYB)	(1-VEN) No	11	SFP full release	2.00E-08
								BURN	VEN NN	12	SFP full release	2.15E-08
				No AC	NA	NA	Yes	HYB NA	NN	13	No release	5.37E-07
		SFP *0.5		S48			(1-B5b2) No Makeup	NA	NA	14	SFP full release	9.83E-07
	SAB	Moderate Break - Fuel Damage and release path assumed					B5b2=.002			15	SFP full release	7.60E-08
	Fuel Damage and release path assumed									16	SFP full release	4.80E-06
											Fuel Uncovery Frequency =	1.74E-05

Figure 8-4. Event Tree for Seismic Initiated Sequences Leading to SFP Fuel Uncovery

Table 8-3 compares different initiating event contributors for the frequency of fuel uncovery at the SFP and for overheating at the ISFSI. These results come from frequencies in Table 4-5. Later in Section 8.4 the frequency of SFP fuel uncovery leading to fuel overheating and release of cesium will also be considered. These frequencies apply equally to all four offload scenarios. The first grey shaded column shows the frequency per year of fuel uncovery caused by seismic events with SNF in the SFP. The frequencies account for the direct impacts of the seismic event leading to fuel uncovery and the indirect effects caused by the seismic failure of the FHB crane dropping a loaded transfer cask into the SFP. It also includes the smaller indirect effects of a seismic initiated severe accident involving one or both of the reactors at DCPP and failure of AC power leading to a radiological release sufficient to limit the ability of plant operators to implement mitigation actions at the SFP.

The probability of a hydrogen burn leading to failure of the FHB (top event HYB in Figure 8-4) was assumed to be 0.5 (USNRC, 2014). No DCPP specific analyses were performed for severe accidents. This mode of failure has negligible impact on the study results because the hydrogen is generated as a result of the SFP fuel overheating. The hydrogen burn may subsequently impact the FHB ventilation system or SFP cooling system but this does not impact the frequency of SFP fuel uncovery leading to the SFP fuel overheating.

The seismic capacity for the FHB ventilation system (assigned a HLPF of 1.49 g SA) was assumed to be the same as the FHB superstructure. This capacity is lower than for the 480v SWGR fans but significantly stronger than for the 480v SWGR ducts. The 480v SWGR ducts are known to be a seismically weak outlier. The FHB ventilation components have not been seismically evaluated. However, the role of the FHV ventilation event in the seismic event tree is not significant. Instead, the analysis uses as a risk measure the probability of a fuel uncovery weighted by the extent of fuel overheating and cesium release to the FHB. The extent of subsequent cesium release from the FHB, which might be partially mitigated by the FHB ventilation system, is not considered. Both sequences with FHB ventilation success and with failure are assigned to the same risk measure.

The frequency in Table 8-3 for SFP fuel uncovery is for the period with one or both of the reactors operating. The periods with and without reactor operation are accounted for as described later during the summation of risks over all periods of time with SNF within the SFPs (from the beginning of 2R22 on May 2, 2021, until all SNF is removed from the SFPs at Unit 2 EOL).

The second grey shaded column presents the frequency contribution for the random drop of a loaded transfer cask (not caused by seismic events) into the SFP resulting in a loss of coolant through a torn liner and failure of the SFP concrete. The modeling approach is to use a conditional probability of 0.1 that the SFP fails in a way to cause a loss of coolant given such a heavy load drop. Two lifts are considered for each MPC transferred to dry storage. One lift is when the transfer cask contains an empty MPC which is placed in the recess area of the SFP. The second lift occurs after the MPC is loaded with fuel assemblies and then lifted above the SFP water for decontamination. The total number of MPCs yet to be transferred to dry storage from both the Unit 1 and Unit 2 SFPs is 80. The vendor option if performed with an MPC capable of handling 37 fuel assemblies could lead to fewer MPCs being transferred. It is also possible that such a newer MPC would be used if one of the other three offload scenarios is selected. Therefore, for purposes of this table all four offload scenarios are evaluated assuming 80 MPCs must be transferred.

Again, these drop frequencies are normalized to frequency of drop per calendar year by dividing the total drop probability by 10 years just for comparison purposes. The total frequency per year of fuel uncovery events involving the SFP is seen as essentially the same as the seismic frequency caused events, the random drop frequencies being negligible by comparison.

The blue shaded column in Table 8-3 represents a total bounding frequency per year that MPCs within storage overpacks already installed at the concrete pads are damaged by events which survive the initiating event screening documented in Appendix A. Aircraft crashes and seismic events much greater than the seismic design basis contribute the most to these frequencies. See Appendix A, Table A-2. The possibility of tornado debris blocking the ventilation ducts for one or more storage overpacks is also assessed, but at a lower frequency.

Each installed storage overpack at the ISFSI pads contains 32 fuel assemblies, or 37 each for the last 80 MPCs if the pre-shutdown vendor option offload scenario is chosen. There are already 58 storage overpacks installed at the ISFSI and up to 80 more are to be installed for a total of 138. So, the number of fuel assemblies present at the ISFSI will range from roughly 10 equivalent reactor cores now to 23 reactor cores once the offloadings to dry storage from both Units 1 and 2 are completed. Both of these values are greater than the maximum seven cores worth of fuel assemblies in one SFP. There are, therefore, more fuel assemblies that may be impacted by the initiating events involving the ISFSI. However, the aircraft crashes, seismic events, and tornado debris are judged unlikely to impact more than several installed storage overpacks in one event. For purposes of comparison, it is judged that on average, five installed storage overpacks would be impacted by an aircraft crash, seismic event, or tornado debris. The number of storage overpacks damaged by a single event, among the 138 present when dry storage is at full capacity, depends on the degree of correlation for the failure modes between the 138 storage overpacks installed at the ISFSI and on the relative heat loads of the storage overpack contents. Some of the storage overpacks contain fuel that was offloaded from the reactor core decades ago. The contents of five storage overpacks are roughly equivalent to just less than the number of fuel assemblies in one reactor core.

Due to the manner in which the event frequencies in Table 8-3 were derived, they are the same for each of the four offload scenarios. The unique aspects of each offload scenario leading to different numbers of fuel assemblies present in the SFP with time and the associated timing of their campaigns to the ISFSI are considered later in a time-step integrated summation of the risks over the time durations that SNF is present in the SFP. The frequency of potential fuel overheating in one or more impacted storage overpacks at the ISFSI is just 7% of the frequency of fuel uncovery at the SFP, which may also potentially lead to fuel overheating. The events affecting the storage overpacks are judged to impact on average about five storage overpacks or less (fewer than one reactor core equivalent of fuel assemblies). There are already storage overpacks present at the ISFSI (58 at this time) and more to be installed from Units 1 and 2 until both SFPs are emptied.

Due to the comparatively small frequency for fuel overheating at the ISFSI as compared to SFP fuel uncovery, and their limited decay heat which limits the potential for fuel overheating, the events affecting the ISFSI are not considered further. Further, the differences in risks at the ISFSI location for the different fuel assembly offload scenarios are expected to be negligible. They were considered up to this point only to provide a basis for the comparison of absolute risks between having fuel in the SFP versus at the ISFSI.

Table 8-3. Comparison of Initiating Event Frequency Contributors Between Potential Fuel Overheating at the SFP and Potential Fuel Overheating while SNF is at the ISFSI Pads

Seismic Frequency per year of Fuel Uncovery while SNF is in Unit 2 SFP (per year)	Random Frequency per year of Dropped HI-TRAC onto SFP causing Fuel Uncovery, 2 lifts per MPC Transferred	Total Frequency of Fuel Uncovery per year at SFP	Frequency of One or More Storage Overpacks With Potential for Fuel Overheating at ISFSI Pad	Ratio of Storage Overpacks Potential Overheating Frequency to SFP Fuel Uncovery Frequency (%)
1.74E-05	3.20E-09	1.74E-05	1.31E-06	7.5%

Figure 8-5 compares the frequencies per year of seismic events leading to fuel uncovery within the SFP and a bounding frequency associated with ISFSI storage overpack overheating with severe accident event frequencies for the Unit 2 reactor while at-power. The first four bars in Figure 8-5 are results from the Unit 2 severe accident assessments performed to date (PG&E, 2019a). The results are presented for two metrics (CDF – core damage frequency, LERF – sequences with the potential for large, early release frequency) and for two sets of initiating events, seismic and the total for all non-seismic.

The non-seismic CDF for the Unit 2 reactor is about twice the Unit 2 reactor seismic contribution. Internal fires contribute most of the non-seismic Unit 2 CDF. The contributions to SFP fuel uncovery from internal fires and internal floods are neglected as they are judged small compared to the seismic contributions. The largest contributors from fire and internal floods to Unit 2 reactor frequencies involved the turbine building, which is not related to the SFP or ISFSI locations. Nevertheless, there are internal fire and flooding events that could take place within the auxiliary building that may impact SFP cooling and the FHB ventilation. However, such initiating events are just not significant by comparison to the seismic fuel SFP uncovery frequency evaluated because of the redundancy and diversity of SFP cooling and makeup options that should be available and are directed by DCPP procedures.

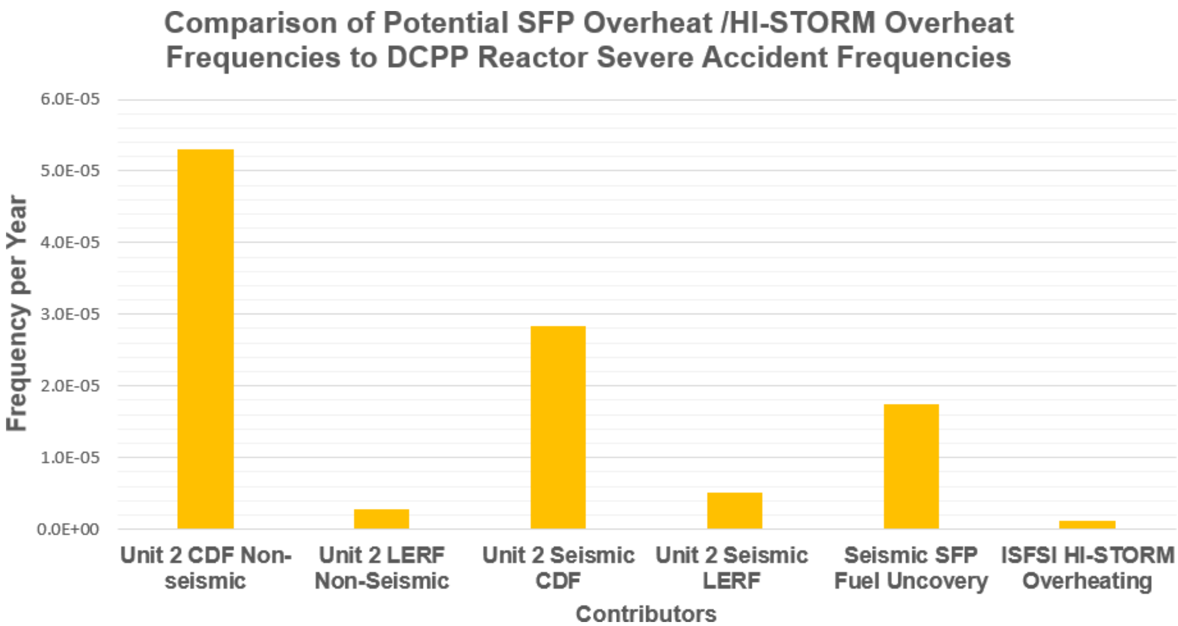


Figure 8-5. Comparisons Between Potential SFP Fuel Overheat and ISFSI Storage Overpack Potential Overheat Frequencies with DCPP Reactor Severe Accident Frequencies

Figure 8-5 shows that the seismic SFP fuel uncovery frequency applicable for all offload scenarios is just 61% of the Unit 2 reactor CDF from seismic events. In a reactor core damage event, fuel overheating is taken as a given for all fuel within the core, whereas the frequency of SFP fuel uncovery listed in Figure 8-4 has not yet considered the extent of fuel overheating given fuel uncovery. However, the Unit 2 reactor potential LERF is lower than the frequency evaluated for seismic SFP fuel uncovery frequency. The term potential contribution to LERF is used because the sequences assigned to LERF are not necessarily capable of producing releases large enough and early enough to result in adverse impacts to public health. Severe accident reactor sequences assigned to LERF for DCPP have in the distant past been assessed as resulting in such consequences, but more recently and with the latest methods and information, the amount of radioactivity released from these same sequences is markedly lower. See Figure 14 of the SOARCA study (USNRC, 2012a). The evaluation of releases from the SFP is not within the scope of this assessment. The extent of fuel judged to overheat given SFP fuel uncovery will be discussed in Subsection 8.4.

It is expected that releases from an SFP fuel uncovery event could occur and be large only if the fuel overheats. There would also likely be time for evacuation prior to any the fuel overheating. If so, such releases would not be considered "Early." They would occur after evacuation is effective. The initiation of SFP fuel uncovery by a large seismic event would complicate evacuation, but the population within 5 miles of the DCPP site is very limited.

As indicated in Figure 8-5, the seismic initiated frequency of SFP fuel uncovery is a substantially greater potential contributor to fuel overheating than that for the ISFSI. Therefore, it is the focus of the remainder of this section.

8.3 Integrated Frequency of Fuel Uncovery Weighted by Equivalent Number of Fuel Assemblies Present

This subsection describes the frequency of SFP uncovery weighted by the effective number of fuel assemblies present in the SFP at the time the seismic event occurs. This subsection does not consider the potential for fuel overheating and cesium release. That is deferred until Section 8.4. By this approach a first measure of the consequences of each fuel uncovery event is considered. Here the effective number of fuel assemblies refers to accounting for the decay of cesium during each fuel assembly's cooling time after reactor offload. The effective number is the assembly count equivalent to the cesium activity within fuel assemblies offloaded to the SFP at just 4 days after reactor shutdown. So if the cesium in one fuel assembly has decayed by half since it was first offloaded from the reactor, that fuel assembly would be counted as half of one fuel assembly. By this approach, fuel assemblies cooled for a long time in the SFP are downgraded by comparison to the fuel assemblies just offloaded from the reactor due to the reduction in amount of cesium available for release. Again, the extent of fuel overheating given SFP fuel uncovery is not considered immediately below, but is to be discussed later in the next subsection.

Table 8-4 first indicates the probability of SFP fuel uncovery due to seismic events while SNF is in the SFP. To obtain this probability, the constant frequency of fuel uncovery is just multiplied by the fuel offload scenario specific duration; i.e., the time interval between the start of refueling 2R22 on May 2, 2021, and the time at which the last fuel assemblies are offloaded from the SFP and transported to the ISFSI for the given offload scenario.

As expected, the vendor option which has the longest remaining duration with fuel in the Unit 2 SFP has the highest integrated probability of SFP fuel uncovery. This is somewhat misleading because in the latter period while some fuel assemblies are in the SFP, the vendor option has more relatively hot fuel assemblies compared to other offload scenarios. It is for this reason that the risk measure described in Section 6 and in this subsection also considers the equivalent number of fuel assemblies present in the SFP at the time the seismic event leads to SFP fuel uncovery.

The grey colored cells in Table 8-4 present the probability of SFP fuel uncovery from seismic events weighted by the effective number of fuel assemblies present, considering cesium decay, at the time of the accident. The weighting by the effective number of fuel assemblies, after considering decay of cesium, is used as a measure for the potential consequences. This risk score will be further adjusted and refined as a risk measure, to account for the potential for fuel overheating, given fuel uncovery as defined in Section 6. This risk measure will be the final basis for comparing the risks of the four offload scenarios. But first intermediate results are considered based only on the effective number of fuel assemblies, after considering decay of cesium. These intermediate results are identified as "risk scores."

Risk score results are shown in Table 8-4 for two time periods in the columns shaded in grey (from May 2, 2021 until Unit 2 shutdown and from May 2, 2021 until all fuel assemblies are removed from the SFP). These results were obtained by a time-dependent integration of the contribution from many time intervals within the two ranges of study dates. While the seismic frequency of SFP fuel uncovery does not vary appreciably with time in the SFP, the effective number of fuel assemblies at risk from the fuel uncovery changes with time and varies depending on the offload scenario followed.

These differences lead to the differences in risk scores seen in Table 8-4. There is a slight variation in the seismic frequency of fuel uncovery due to the operation of the adjacent reactors. This slight variation in uncovery frequency is accounted for with time by this assessment. The post shutdown 7-year offload scenario has the highest integrated probabilities weighted by effective fuel assemblies present for both time intervals. Relative to the post shutdown 7-year offload scenario, the other scenarios have lower risk scores. The blue shaded column indicates the percentages of the post shutdown 7-year offload scenario for each of the other offload scenarios. Although each offload scenario has the same seismic initiated frequency of fuel uncovery, the pre-shutdown vendor option offload scenario has only 40% of the integrated probability weighted by the effective number of fuel assemblies present. The pre-shutdown vendor offload scenario risk score also has about half the risk score of the other two pre-shutdown offload scenarios. As expected, the pre-shutdown 5-year offload has a slightly lower risk score than does the pre-shutdown, 7-year offload scenario. Table 8-4 shows a distinct difference in the risk score for the four offload scenarios when the equivalent number of fuel assemblies present at the postulated time of the accident is taken into account.

Table 8-4. Time Integrated Probabilities of SFP Fuel Uncovery Weighted by Effective Number of Fuel Assemblies Present Considering Cesium Decay

Offloading Scenarios (Years refer to SFP empty time after Unit 2 EOL)	Integrated Probability of SFP Uncovery (Not weighted by Number of Fuel Assemblies Present)	Seismic SFP Probability of Fuel Uncovery until Unit 2 EOL (2025), weighted by effective # of FAs in SFP	Seismic SFP Total Probability of Fuel Uncovery Until Empty, weighted by effective # of FAs in SFP	Ratio of Weighted Total Fuel Uncovery Probabilities to Post-SD 7-year offload(%)
Post-shutdown 7-year offload	1.97E-04	0.057	0.147	100%
Pre-shutdown 7-year offload	1.97E-04	0.049	0.126	86%
Pre-shutdown 5-year offload	1.62E-04	0.049	0.106	72%
Pre-shutdown vendor option offload	2.20E-04	0.027	0.057	39%

Figure 8-6 provides a more detailed look at the integrated probabilities of fuel uncovery in the SFP as weighted by the number of equivalent fuel assemblies present in the SFP at the time of the seismic event, adjusted for cesium decay. Here the weighted cumulative probabilities are shown as a function of time spent after 2R22 with SNF in the SFP. The far-right points in these curves match the integrated and weighted probabilities in Table 8-4. Recall that the Table 8-4 results integrated until the SFP is emptied are cumulated over all times in the study period. The lower slope of the pre-shutdown vendor option reflects the significant reduction in fuel assemblies for this offload scenario compared to the other three. The blue curve, compared to the yellow curve, indicates the benefits of completely moving the fuel to dry storage within 5 years versus 7 years after Unit 2 permanent shutdown. Recall that the final shutdown of Unit 2 is scheduled for late August of 2025. The pre-shutdown vendor option is seen to lead to lower cumulative probabilities both before and after the Unit 2 final shutdown. This reflects the more extensive pre-shutdown campaigns to transfer fuel assembles to the ISFSI. The small wiggles in the curves are due to uneven discretization of the time intervals and does not reflect any specific loading or offload events.

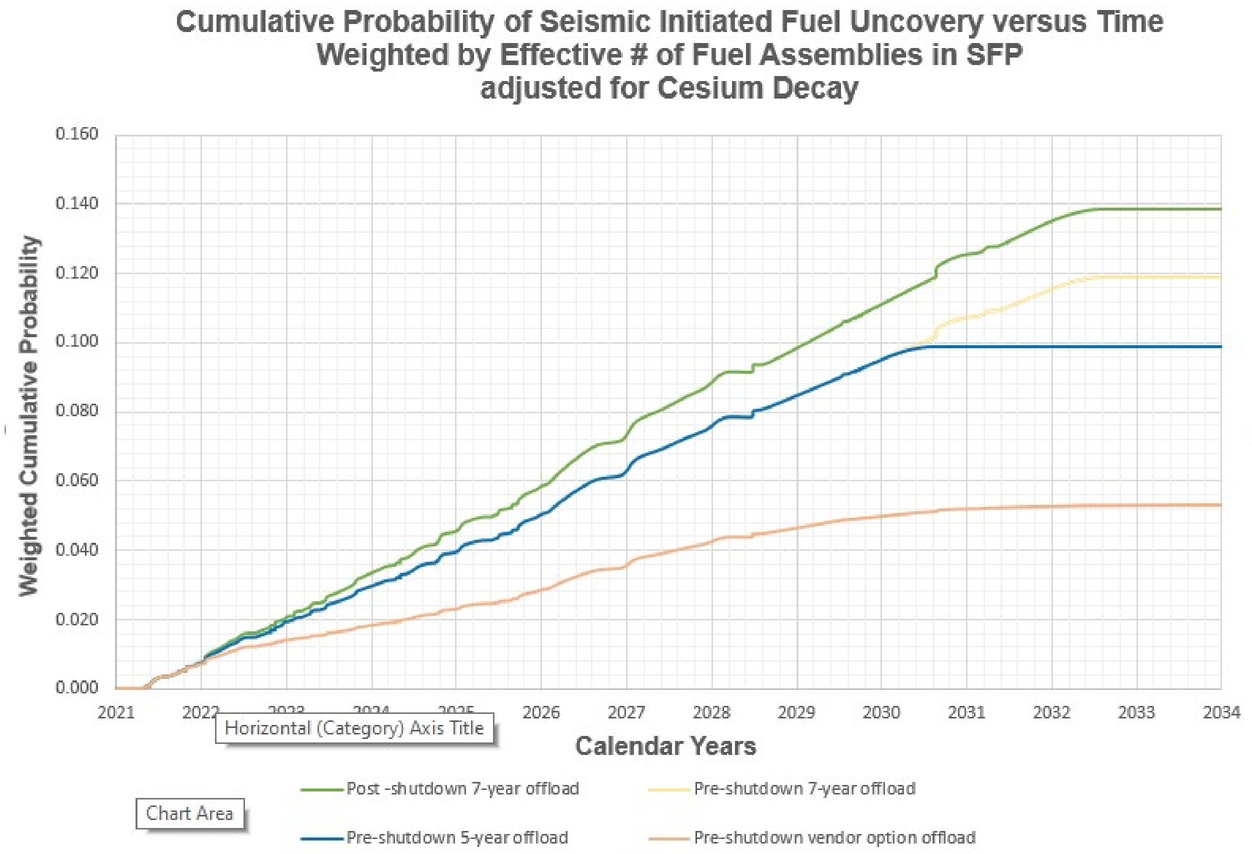


Figure 8-6. Cumulative Probability of Seismic Initiated Fuel Uncovery versus Time Weighted by the Number of Fuel Assemblies in the SFP Adjusted for Cesium Decay

Figure 8-7 shows another way to see the variation in risk for different ranges of fuel assemblies present in the SFP. This figure is similar to Figure 8-2, which represents the time spent with different amounts of fuel in the SFP, except this time the vertical axis durations in years are scaled by the frequency per year of seismic initiated fuel uncovery in the SFP. The result is a plot of the probability of seismic SFP uncovery versus number of equivalent cores of fuel assemblies present in the SFP at the time of the fuel uncovery and for each of the four offload scenarios after adjusting for cesium decay. It is again observed that the pre-shutdown vendor option offload scenario (shown with orange bars) results in the largest probabilities of a seismic initiated SFP fuel uncovery for configurations involving less than three reactor core equivalents of fuel assemblies in the SFP. By contrast, the other three offload scenarios have very low probabilities with this low-end range of fuel assemblies in the SFP. The post-shutdown 7-year offload scenario (shown in green) is the only option among the four which has a probability of the SFP fuel uncovery event occurring with greater than six equivalent cores of fuel assemblies in the SFP. The other two offload scenarios (pre-shutdown 7-year in yellow and pre-shutdown 5-year in blue) have their highest probabilities of SFP fuel uncovery occurring with five to six equivalent cores of fuel assemblies in the SFP.

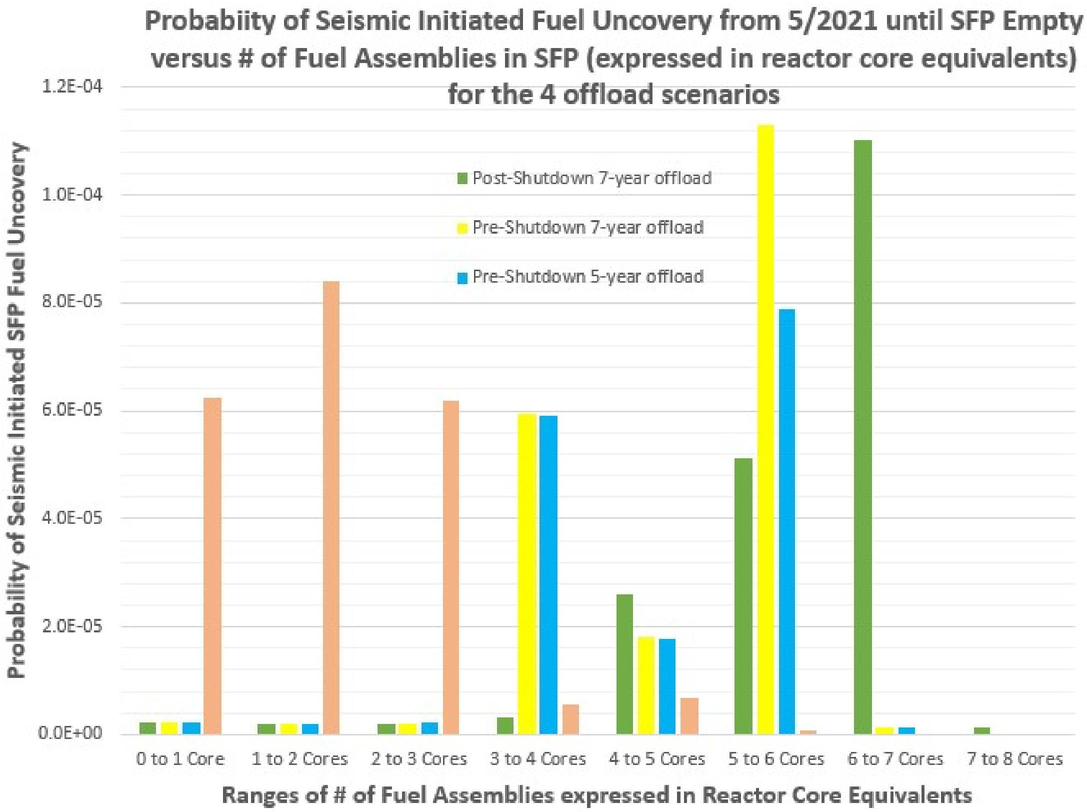


Figure 8-7. Probability of Seismic Initiated SFP Fuel Uncovery Versus Number of Equivalent Reactor Cores of Fuel Assemblies

Figure 8-8 examines the contributors to seismic SFP fuel uncovery by ranges of seismic accelerations expressed in units of g, spectral accelerations (SA). Each vertical bar is made up of two parts. The blue portion represents the contribution from direct seismic impacts. The orange portion of the bars represents the added frequency that comes from a severe accident in the operating Unit 2 reactor resulting in enough radiological release to the environment to make mitigating actions for the SFP untenable. The probability for a concurrent severe accident in an operating reactor at-power conditions in each time interval comes from the assessed seismic frequency of a large, early release (Figure 8-4). This same frequency is applied for all conditions involving the Unit 2 SFP, both for refueling and for non-refueling periods. When Unit 2 is in a refueling outage, the Unit 1 reactor is assumed at-power so that the same frequency applies. The reactor sequences involving a large early release are judged to result in sufficient radiation onsite and could interfere with operator actions outside the control room. Core damage sequences in which the reactor containment remains leak tight should not release sufficient radiation onsite to impact the performance of the operators attempting to respond to the impacted SFP.

For this figure, the added frequency from a Unit 2 reactor severe accident resulting in LERF includes both the contributions from SFP fuel uncovery sequences involving a loss of all AC (sequences 6 and 13) and from those sequences which do not fail all AC (sequences 1, 2, 3, 8, and 9). See Figure 8-8 for the presentation of these seismic severe accident sequences within the SFP seismic event tree. The total SFP fuel uncovery frequency per year includes both the direct and the added contribution due to LERF level releases from a reactor core damage event.

Since the y-axis is logarithmic, the main contributors to the frequency are from accelerations in the range of 3.0 to 9.0 g SA. Next in importance are in the range greater than 2.0 g up to 3.0 g SA. The DCPP seismic bounding design basis from the Hosgri fault is at 2.1 g SA. The auxiliary building seismic capacity has a high confidence of low probability of failure (HCLPF) at 2.33 g SA. The seismic capacity of the SFP structure and liner has an HCLPF of 3.5 g SA as evaluated by PG&E. The structural seismic capacities explain the increases in seismic initiated SFP fuel uncovery above 2.0 g SA.

The jump in frequency in the range of 3.0 to 3.5 g SA is not explained by these structural seismic capacities. Instead, human error probabilities were increased to 1.0 for all accelerations greater than 3.0 g SA. For seismic accelerations greater than 3.0 g SA, the likelihood of core damage at one or both reactors if they are operating is assessed as very likely (about 95%) and all AC power has failed. See Assumption 4 in Appendix B for additional discussion of this point. Since offsite power is very likely to be unavailable for such strong seismic events, and that would disable the SFP cooling pumps, eventually some operator actions are then required to mitigate fuel uncovery, even in the absence of other seismic equipment failures. This approach to accounting for the increased stresses and equipment failures when considering the human error probabilities following strong seismic events is consistent with the approach taken for seismic risk evaluations of the Unit 2 reactor core. The width of the accelerations is widened for the highest two ranges, or the contribution of the highest four ranges would drop off in a monotonically decreasing fashion, reflecting the decreases in the seismic hazard frequency for such high accelerations.

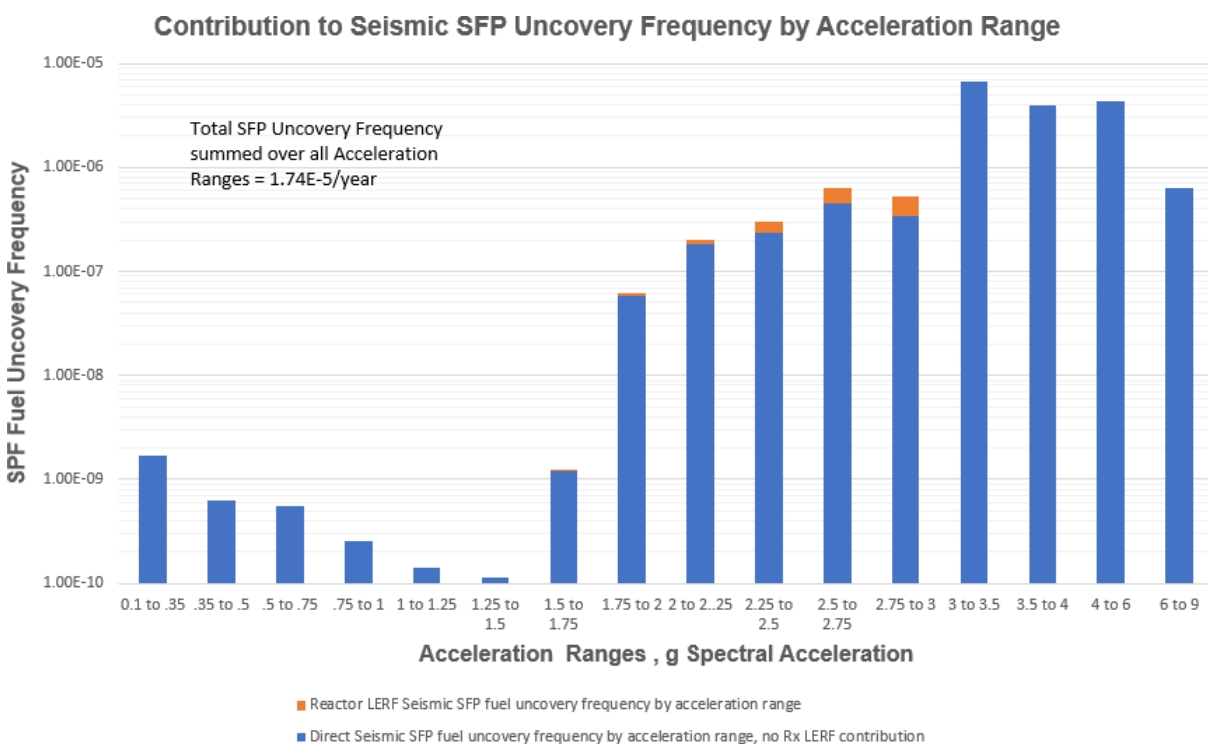


Figure 8-8. Contribution to Seismic SFP Uncovery Frequency by Acceleration Range

Turning up the microscope further on contributors, Figure 8-9 presents the seismic event tree response for the acceleration range with the highest frequency of fuel uncovery (3.0 to 3.5 g SA). The highest frequency sequence contributing to the frequency of seismic initiated SFP fuel uncovery for this range is highlighted in red font (Sequence 7). Sequence 7 involves the occurrence of the seismic event, the successful response of the fuel handling building, SFP, and SFP small leaks through the liner, but both offsite and onsite AC power are unavailable. Because the acceleration is greater than 3.0 g SA, a 1.0 value is used for all human error probabilities, precluding credit for the implementation of SFP makeup using FLEX. For seismic accelerations greater than 3.0 g SA, the likelihood of core damage at one or both reactors, if they are operating, is assessed as very likely (about 95%) and all AC power has also failed. See Assumption 4 in Appendix B for additional discussion of this point. Since offsite power is very likely to be unavailable for such strong seismic events and that would disable the SFP cooling pumps, some operator actions are then required to mitigate fuel uncovery even in the absence of other seismic equipment failures. This approach to accounting for the increased stresses and equipment failures when considering the human error probabilities following strong seismic events is consistent with the approach taken for seismic risk evaluations of the Unit 2 reactor core for at-power conditions. Other sequences through the event tree for this acceleration range involve other equipment and structural failures. However, each of these sequences also leads to SFP fuel uncovery, so that the total for all paths resulting in SFP fuel uncovery matches the initiating event frequency. Sequence 7 just happened to have the highest frequency among all of those individual sequences initiated by a seismic event in the range 3.0 to 3.5 g SA resulting in SFP fuel uncovery.

Initiator	FHB/ AUXB Intact 2.3g HCLPF	SFP Intact, No Moderate Breaks (1) 3.54g HCLPF	SFP Intact, No Small Breaks 3.54g HCLPF	AC Power Available <2.06g HCLPF	SFPC SFP Cooling HCLPF=2.50g	NEM Normal and Emerg.. Makeup	B5b FLEX Makeup	HYB No H2 Burn	FHB Ventilation HCLPF = 1.49g	Sequence ID	Consequence	Sequnce Frequencies for SEIS13 (per calendar year)	Comment
Seimsic 3.0-3.5g SA	Yes	Yes	Yes	Yes	Yes	NN	NN	NA	NN	1	No release	6.94E-10	SFP cooling continues
6.68E-6 per cal. year	(1-SAB)	(1-SFP*0.5)	(1-SML)	(1-S48)	(1-SFPC) No	Yes	NN	NA	NN	2	No release	0.00E+00	Makeup maintains FA cooling
					SFPC	(1-NEM1) No	Yes	NA	NN	3	No release	0.00E+00	Either source of makeup covers fuel
						NEM1=1.0	(1-B5b1) No Makeup	NA	Yes	4	Filtered SFP release	0.00E+00	No burn if boil-off sweeps building of air
							B5b1=1.0		(1-VEN) No	5	SFP full release	5.15E-07	No SFPC or makeup
				No AC	NA	NA	Yes	NA	VEN=1.0 NA	6	No release	0.00E+00	AC required for cooling and normal makeup
				S48 - .907			B5b2= 1.0 No Makeup	NA	NA	7	SFP full release	5.01E-06	No burn if boil-off sweeps building of air, no filtering due to loss of AC power
			Smaller Leak Drains to bottom of pool	Yes	NA	Yes	B5b2= .002 NN	NA	NN	8	No release	0.00E+00	Draining precludes SFP cooling
			SML=SFP *0.5	(1-S48)		(1-NEM2) No makeup	Yes	NA	NN	9	No release	0.00E+00	
						NEM2=1.0	(1-B5b1) No Makeup	No BURN	Yes	10	Filtered SFP release	0.00E+00	Much time available for makeup alternatives
							B5b1=1.0	(1-HYB)	(1-VEN) No	11	SFP full release	1.81E-08	
								BURN	VEN=1.0 NN	12	SFP full release	1.81E-08	Burn assumed to fail FHB integrity
				No AC	NA	NA	Yes	HYB=1.0	NN	13	No release	0.00E+00	AC required for cooling and normal makeup
		SFP=5.77E-3		S48=.907			(1-B5b2) No Makeup	NA	NA	14	SFP full release	3.52E-07	Loss of AC assumed to fail AUXB integrity
	SAB=.112	Moderate Break - Fuel Damage and release path assumed					B5b2=.002			15	SFP full release	8.56E-09	SFP structural failure resulting in moderate break size; no credit for SFP makeup (8)
	Fuel Damage and release path assumed									16	SFP full release	7.48E-07	FHB failure precludes all SFP cooling and makeup
											Fuel Uncovery Frequency =	6.68E-06	

Figure 8-9. Key Sequence for Seismic SFP Fuel Uncovery, from 3.0 to 3.5 g SA Range

Figure 8-10 presents the results of sensitivity evaluations for the impacts of alternate assumptions on just the seismic initiated SFP fuel uncovery frequency. The base case frequency is 1.74E-5 per year. The additional contributions are unknown from considering the impacts of radiological releases onsite from a severe accident in one of the operating reactors which may occur early enough and be large enough to prevent operator actions at the SFPs. The blue colored bars represent the direct seismic impacts leading to SFP fuel uncovery. The orange bars indicate the added radiological release contributions. The sum of the two bars yields the total SFP fuel uncovery frequency due to seismic impacts when at least one of the DCPP reactors is operating. For the duration that fuel assemblies are still in the Unit 2 SFP but neither reactor unit is operating, the SFP fuel uncovery frequencies associated with just the blue colored bars apply (the orange colored contributions do not apply.)

Removing the two seismic structural failures which directly lead to SFP fuel uncovery (SFP and the auxiliary building) leads to a modest 4% reduction in the SFP fuel uncovery frequency. When reducing only the human error probabilities for high accelerations (those error probabilities assigned for accelerations greater than 3.0 g SA) to the same human error probabilities used at lower accelerations, a much larger frequency reduction is seen. The revised seismic initiated SFP fuel uncovery frequency becomes 8.9E-6 per year, which is a 49% decrease. It is seen that by eliminating direct failure impacts the contribution caused by radiation impacts from a concurrent reactor core damage increases substantially. This contribution overrides the otherwise assumed low human error probabilities (HEP).

The third assumption set combines the changes from the first two alternate assumption sets; i.e., low HEPs and removal of the SFP and auxiliary building seismic failures. As expected, the total SFP fuel uncovery then reduces to just 5.6E-6 per year, and again the contribution from radiation impacts from a concurrent reactor core damage event increases.

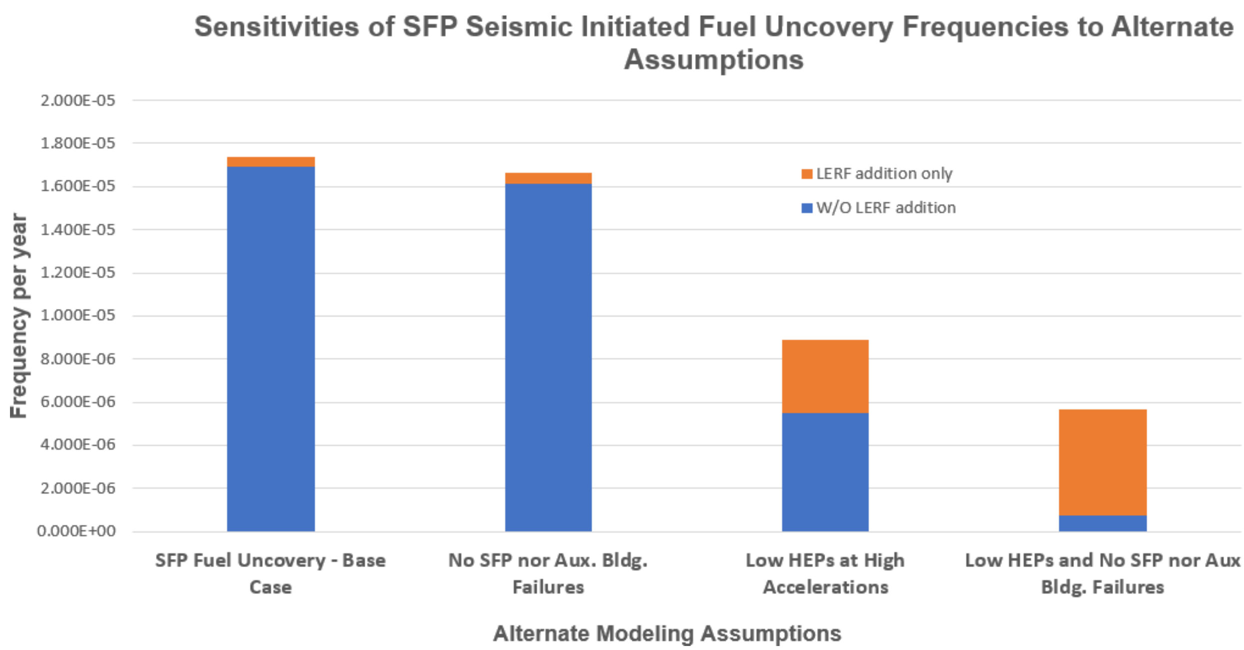


Figure 8-10. Sensitivities of SFP Seismic Initiated Fuel Uncovery Frequency to Alternate Modeling Assumptions

Unfortunately, it would be difficult to justify low human error probabilities following such large seismic events. One alternative interpretation would be if the local operator actions represented were not required in order to avoid radiological release from the uncovered fuel in the SFP. The actions credited in the current sequence representations only apply if the SFP and auxiliary building structures remain intact so that the SFP equipment such as pipes and valves are available, and the geometry is maintained. If the encompassing auxiliary building or the SFP structure itself fails from seismic acceleration, the current model does not credit any local operator actions due to the difficulty such structural failures would impose. This assumption of not crediting local operator actions when one or both of the key structures fail is an important one and not related to the above discussions of increasing the human error probabilities for accelerations greater than 3.0 g SA. The actions credited in the current seismic sequence representations only apply if the SFP and auxiliary building structures remain intact so that the SFP equipment such as pipes and valves are available, and the geometry is maintained. If it could be shown that natural convection air cooling could limit fuel cladding temperatures in response to a loss of SFP cooling or for smaller leaks in the SFP liner without local operator actions, then the 49% reduction in seismic SFP fuel uncovery frequency seen in Figure 8-10 would apply. This point is worth further discussion, but in relation to evaluating the extent of fuel overheating given fuel uncovery. This is discussed below.

8.4 Probability of Fuel Uncovery Weighted by the Effective Number of Fuel Assemblies Present that Overheat and Release Cesium

The analyses presented in Section 8.3, including the sensitivities presented in Figure 8-10, are based on (1) the frequency per year of seismic initiated fuel uncovery events involving the SFP and (2) the integrated probability of a seismic initiated SFP uncovery event during the time that fuel assemblies are present in the SFP, weighted by the effective number of fuel assemblies present at each interval in time accounting for cesium decay. The risk scores evaluated for each offload scenario in Section 8.3 do not consider the extent of fuel overheating and therefore the amount of cesium actually released from the fuel assemblies. The analysis in this section extends the Section 8.3 analyses to also address the extent of fuel overheating leading to cesium release. Here the risks of cesium release are developed for fuel assemblies only while they are in the SFP. The contributions to risk from fuel assemblies overheating while in transport or while in storage at the ISFSI have already been shown to be relatively small as compared to fuel uncovery risk (see Tables 4-7 and 8-3).

This section presents a number of sensitivity results for alternate assumptions about the extent of fuel overheating, given fuel uncovery in the SFP. These sensitivity cases were selected following a literature review documented in Appendix C. The cases were evaluated to provide insights on the controlling assumptions about the differences in resulting risk measures between the four offload scenarios. After the sensitivity results are presented and discussed, the base case assumptions for the extent of fuel overheating given fuel uncovery are also developed. The base case risk measures for each of the four offload scenarios accounting for the extent of fuel overheating are then developed.

The number of fuel assemblies present in the SFP varies with time due to the four reactor offloads and the timing of the campaigns transferring fuel assemblies to the ISFSI. The timing of the campaigns is offload scenario dependent. In each of the risk score results presented in Section 8.3, the number of fuel assemblies present is scaled down by the amount of cesium remaining after decay as a function of time following each reactor offload. For the risk scores and for the risk measures developed in this section, this reduction to account for cesium decay is evaluated separately for each reactor offload, including for reactor offloads prior to the calendar years studied beginning in 2021 (2R22). When the phrase "effective number of fuel assemblies present at each time" is used, it should be understood that this means that the decay of cesium within the fuel assemblies while in the SFP has been considered. Cesium is considered a useful measure of the fuel assembly inventory. It is described in Section 6 that cesium releases caused by fuel overheating are expected to lead to the greatest differences in public health risks as evaluated for the four different offload scenarios considered.

What was not incorporated into the risk scores presented in Section 8.3 is an assessment of the extent of fuel overheating, and consequently the expected amount of cesium released given an event involving

SFP fuel uncovery. This is a difficult issue since as noted at the end of Section 5, there are no integrated models that have been validated to compute the extent of fuel overheating. The treatment of this issue is described in the remainder of this subsection.

Results of a set of sensitivity studies that considered the extent of fuel overheating and cesium release are presented in Figure 8-11. Figure 8-11 examines the possible bases for evaluating the extent of fuel overheating given a fuel uncovery event. The risk measure defined in Section 6 requires the incorporation of the extent of fuel that overheats. The risk measure is then used to perform the final comparison of the risks of the four offload scenarios. What makes this risk measure different from the previously described risk score is that it also imposes new assumptions on the fuel assemblies which participate in overheating, given an accident resulting in fuel uncovery. Each case listed on the x-axis of Figure 8-11 considers an alternate assumption set which is to be applied in the evaluation of the risk measure. The assumption sets are explained below starting from the left side of the x-axis of the figure, along with observations of their results.

The first sensitivity case labeled “60 days” in Figure 8-11, given a fuel uncovery sequence, restricts the participation of fuel assemblies from overheating to those just offloaded from the reactor within the last 60 days. These results were obtained by a time-dependent integration of the contribution from the time intervals covering the first 60 days after each reactor offload. While the seismic frequency of SFP fuel uncovery does not vary with time in the SFP during this 60-days, the effective number of fuel assemblies at risk does vary with time depending on the offload scenario followed. These differences lead to the very small differences in risk scores presented in Figure 8-11 for this case. Fuel assemblies offloaded from the reactor in an earlier refueling are assumed not to overheat. For the four upcoming reactor offloads included in this assessment, there are four periods of 60 days each immediately following the offload of fuel from the reactor. Offloaded fuel assemblies have their highest heat load during the first 60 days. A typical refueling outage is planned for only 33 days. Within the first 60 days of a reactor offload, PG&E has committed to arrange the offloaded fuel assemblies so that the highest decay power assemblies are all surrounded by relatively low power fuel assemblies (the 1x4 arrangement pictured in Figure 5-1). In practice, PG&E tries to complete the 1x4 arrangement before the refueling outage is over, time permitting. The 1x4 arrangement ensures that the time to 900°C (the temperature at which Zr-air reactions may be self- sustaining) is at least 10 hours following the fuel uncovery. The time any given SFP fuel uncovery sequence takes to lose its coolant (to below the top of the active fuel) would be added to this 10-hour time.

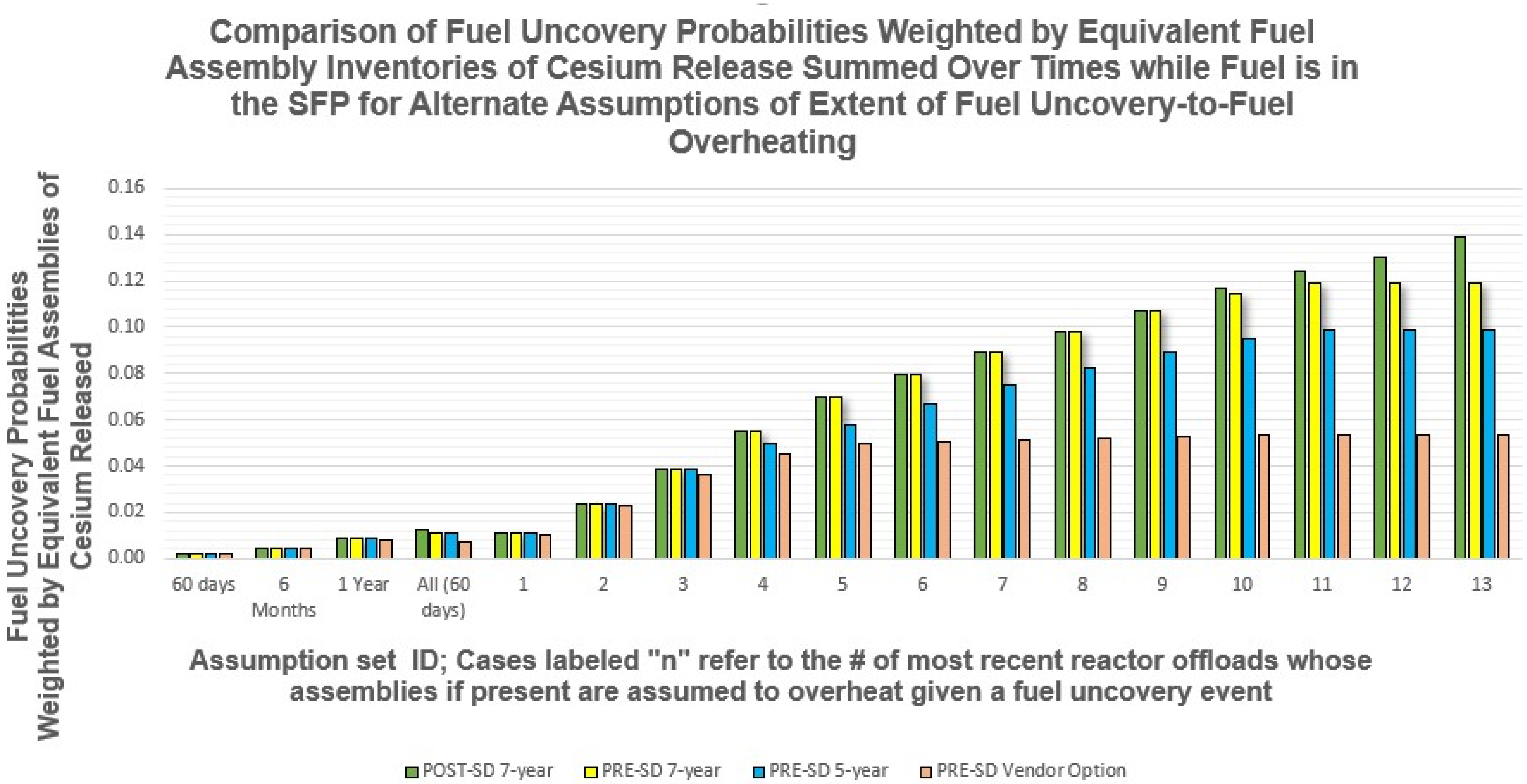


Figure 8-11. Comparison of Fuel Uncovery Probabilities Weighted by Equivalent Assembly Inventories of Cesium Release Summed Over Time while Fuel is in the SFP and Compared for Alternate Assumptions of Extent of Fuel Overheating Given Fuel Uncovery

The 10-hour time for heatup is judged to give operators sufficient time to take local actions that will mitigate the loss of SFP coolant.

The assumptions which make up the “60 days” case are optimistic in that it is further assumed that all fuel assemblies offloaded in prior refueling outages have sufficiently low heat loads that they would not reach the 900°C, let alone overheat the fuel. The previously offloaded fuel assemblies have all been cooling in the SFP for at least an entire refueling cycle of approximately 18 months. Since they do not overheat, their cesium inventory is excluded from the amount assessed as being released from the SFP. This assumption set is slightly pessimistic in that all fuel assemblies recently offloaded are assumed to overheat, even though some fuel assemblies have noticeably lower burnups and power peaking factors than average. As expected, the results for all four offload scenarios are similar for this “60 days” case.

The case labeled “6 months” in Figure 8-11, given a fuel uncovery, limits the fuel assemblies that overheat to those offloaded from the reactor within the previous 6 months. This assumption greatly increases the cooling time during which fuel assemblies are still assumed to exceed the 900°C temperature limit. Again, fuel assemblies offloaded from the reactor in an earlier refueling are assumed not to overheat. For the four upcoming reactor offloads included in this assessment, this corresponds to four periods of 6 months each immediately following the offload of fuel assemblies when fuel may overheat. This “6 months” case assumption requires a much longer period for cooling of the fuel assemblies before it is assumed that they would not reach the 900°C temperature limit given fuel uncovery. This case is optimistic in that it is further assumed that all fuel assemblies offloaded in prior refueling outages have sufficiently low heat loads that they would not reach the 900°C, let alone overheat the fuel. The previously offloaded fuel has been cooling in the SFP for at least an entire refueling cycle of approximately 18 months. If the fuel uncovery occurs towards the end of the 6-month period, previously offloaded fuel would have been cooling for 24 months. Since these assemblies do not overheat, the cesium inventory they contain is excluded from the amount assessed to be released from the SFP. This “6 months” case is again slightly pessimistic in that all fuel assemblies recently offloaded are assumed to overheat, even though some fuel assemblies have noticeably lower burnups and power peaking factors than average and so would have lower heat loads than the average fuel assembly. As expected, the results for all four offload scenarios are again similar for this case. Recall that fuel assemblies are not transferred to the ISFSI during the first 6 months following a reactor offload. If the offload scenario includes campaigns to the ISFSI before EOL, the cooler fuel assemblies, with more time spent cooling in the SFP are instead transferred.

The case labeled “1 year” in Figure 8-11, restricts the participation of fuel assemblies from overheating to those just offloaded from the reactor within the last year. This assumption further increases the cooling time during which fuel assemblies are assessed to exceed the 900°C temperature limit if fuel uncovery occurs. Again, fuel assemblies offloaded from the reactor in an earlier refueling are assumed not to overheat even if the SFP fuel is uncovered. For the four upcoming reactor offloads included in this assessment, this corresponds to four periods of 1 year each immediately following the four offloads of fuel. This period requires a much longer time for decay of the fuel assemblies than those described previously, before it is assessed that they would not reach the 900°C temperature limit. This case is still optimistic in that it is further assumed that all fuel assemblies offloaded in prior refueling outages have sufficiently low decay power that they would not reach 900°C let alone overheat the fuel. The prior offloads of fuel assemblies have all been cooling in the SFP for at least an entire refueling cycle of approximately 18 months. If the fuel uncovery occurs towards the end of the 1-year period, the previously offloaded fuel would have been cooling for 30 months. Since the previously offloaded fuel assemblies do not overheat, the cesium inventory they contain is again excluded from the amount found to be released from the SFP. It is slightly pessimistic in that all fuel assemblies recently offloaded are assumed to overheat, even though some fuel assemblies have noticeably lower burnups and power peaking factors than average and so would have lower heat loads than the average fuel assembly. As expected, the results for all four offload scenarios are similar for this case. Recall that the most recently offloaded fuel assemblies are not transferred to the ISFSI during the first year. If the offload scenario includes campaigns to the ISFSI before EOL, the cooler fuel assemblies with more time spent cooling in the SFP are instead transferred.

The case labeled “All (60-days)” in Figure 8-11 is similar to the first case labeled “60-days”. This case also restricts the periods that fuel assemblies are assumed to overheat to just the 60-day period following the most recently offloaded fuel. However, in this case, the fuel assemblies recently offloaded are assumed to overheat and this time to propagate to all other fuel assemblies present in the SFP, including those cooled for many years, causing them to also overheat. The full extent of propagation assumed in this case could be caused by the exothermic reactions of a cladding fire, or just as a result of the total SFP heat load being very high immediately following the reactor offload. For the four upcoming reactor offloads included in this assessment, there are four periods of 60 days each immediately following the offload of fuel from the reactor. This case is pessimistic in that it assumes all fuel assemblies present are involved in overheating, but optimistic in that it excludes the possibility of fuel assemblies recently offloaded from overheating at all after they have cooled more than 60 days after the reactor offload. Some differences between the offload scenarios are seen for this case. These differences reflect the fuel assembly offloads in prior fuel cycles which differ between offload scenarios.

The remaining cases in Figure 8-11 labeled “1” to “13” have similar meanings so they are explained together. This set of cases restricts the participation of fuel assemblies from overheating to those just offloaded from the reactor during the previous “n” reactor offloads.

If n=1, then only fuel offloaded in the most recent refueling outage is vulnerable to overheating. The other fuel assemblies offloaded in prior refueling outages have all cooled for more than 18 months. Unlike for the “1-year” case, the fuel assemblies most recently offloaded are assumed vulnerable for the entire refueling cycle, about 1.5 years after first being offloaded. For the four upcoming reactor offloads included in this assessment, there is fuel assumed vulnerable to overheating throughout the refueling cycle; i.e., back to back with each refueling offload. After Unit 2 shuts down, the most recently offloaded fuel assemblies are assessed as being vulnerable to overheating for the next 18 months. This case is optimistic in that it is further assumed that all fuel assemblies offloaded in prior refueling outages have sufficiently low heat loads that they would not reach 900°C and not overheat any fuel. All the fuel assemblies offloaded in earlier refueling outages have been cooling in the SFP for at least an entire refueling cycle of approximately 18 months. If the fuel uncovery occurs towards the end of the current refueling cycle, all previously offloaded fuel would have been cooling for at least 36 months. Since these previously offloaded assemblies are assumed not to overheat, their cesium inventory is excluded from the amount found to be released from the SFP. This case is slightly pessimistic in that all fuel assemblies most recently offloaded are assumed to overheat, even though some fuel assemblies have noticeably lower burnups and power peaking factors than average and thus would have lower decay power than the average fuel assembly. Since fuel assemblies from prior reactor offloads are not vulnerable to overheating, the results from this case look very similar to those from the “1-year” case. There is not much difference between the results for the four offload scenarios.

For the sensitivity case with n=2, only fuel offloaded in the previous and most recent refueling outage is assessed as overheating given a fuel uncovery. None of the other fuel assemblies offloaded in still earlier refueling outages prior to the most recent two is assessed as vulnerable to fuel overheating. The fuel assemblies most recently offloaded are assumed vulnerable for the most recent refueling cycle, plus a second refueling cycle when a second set of fuel assemblies are offloaded (for a period of 3 years after first being offloaded from the reactor). For the four upcoming reactor offloads included in this assessment, there is fuel vulnerable to overheating throughout the refueling cycles; i.e., back to back with each refueling offload. In this sensitivity case, the fuel assemblies most recently offloaded and those in the prior refueling outage are both assumed vulnerable to overheating; i.e., n=2 for two refueling offloads. The fuel assemblies deemed vulnerable to overheating changes each refueling cycle. After two refueling cycles, the fuel assemblies are no longer assessed as vulnerable to fuel overheating given a fuel uncovery event. After Unit 2 shuts down permanently, the vulnerable fuel assemblies persist for another two refueling cycles, or about 3 years.

The assumptions which make up this case are optimistic in that it is further assumed that all fuel assemblies offloaded prior to the last two refueling outages have sufficiently low heat loads that they would not reach 900°C and not overheat the fuel. Since these previously offloaded assemblies do not overheat, their cesium inventory is excluded from the amount found to be released from the SFP. This assumption set is slightly pessimistic in that all fuel assemblies recently offloaded and from the prior reactor offload are assumed to overheat, even though some fuel assemblies have noticeably lower burnups and power peaking factors than average and would have lower heat loads than the average fuel assembly. Again, the results for this case look very similar to those from the "n=1" reactor offload sensitivity case. That is, there is not much difference between results for the four offload scenarios.

The above discussion for n=2 is similar for n=3 to n=13. The differences in these cases only reflect the number of prior reactor offloads of fuel assemblies that are assessed as vulnerable to fuel overheating, given a fuel uncovery event occurs. As "n" increases, the time of Unit 2 reactor shutdown does not change. The period of time after Unit 2 permanent shutdown that some fuel assemblies are still assessed as vulnerable does increase with "n". If n=4, the fuel in the SFP is assumed vulnerable for the length of four fuel refueling cycles, or about 6 years. More importantly, for increasing "n" the assumption is that the fuel assemblies from the previous reactor offloads that are considered vulnerable to overheating are increasing. This is true for the number of fuel assemblies both prior to Unit 2 permanent shutdown and after permanent shutdown. The four defined offload scenarios considered in this assessment were developed after determining that the scheduled number of fuel assemblies transferred with time are eligible for transfer to the ISFSI considering the restrictions on total heat load in each MPC.

One caveat for all these sensitivity cases is that to be vulnerable to fuel overheating, the fuel assemblies have to be present in the SFP at the time of the SFP fuel uncovery event. The campaigns to the ISFSI varies in timing and number of MPCs transferred for each offload scenario. The modeling assumption is that the fuel assemblies offloaded first in these campaigns are those which have cooled the longest in the SFP. It is recognized that this modeling assumption that the first fuel assemblies offloaded from the reactor are also those first transferred to the ISFSI is an approximation to plant practices. If relatively hotter fuel assemblies were instead transferred to the ISFSI first, then some reduction in the assessed potential for cesium release given a fuel uncovery event would occur. Such reductions in risk are judged to be minor, however, because only fuel that has cooled for at least 5 years in the SFP is actually transferrable to the ISFSI. This is true for all four offload scenarios evaluated in this assessment.

Plant practice is to transfer a collection of fuel assemblies whose total heat load is as close to the MPC rated design heat load as possible. Therefore, the initial heat load of each MPC transferred to the ISFSI is modeled as being equal to the design heat load for an MPC. The ISFSI FSAR (PG&E, 2018a) makes this same assumption for all safety analyses involving the ISFSI. Decay of the total individual MPC heat loads while in storage at the ISFSI would also reduce the risk of cesium release at the ISFSI. This reduction was also neglected for the bounding assessment of ISFSI storage risks as described in Section 8.1 and Appendix A

The results illustrated in Figure 8-11 reveal some insights into what assumptions may impact the risk measure for comparing the offload scenarios. Recall that the ordinate of Figure 8-11 presents for each case the four offload scenario risk measures that consider all of the following:

1. The seismic initiated frequency of core uncovery is integrated over times since the beginning of refueling outage 2R22 until all fuel is transferred from the SFP to the ISFSI. These time durations vary with the specific offload scenario. Although they contribute negligibly, the random and seismic initiated drop probabilities of a loaded transfer cask onto the SFP leading to a loss of SFP coolant and fuel uncovery are also considered. These discrete events are matched up with the date that each loaded MPC is transferred to the ISFSI for the offload scenario being evaluated.
2. At each point in these time durations when a seismic event occurs, the measure is weighted by the equivalent number of fuel assemblies present at that point. They are equivalent in that they are adjusted for the decay of cesium while located in the SFP. This decay in the amount of cesium present is performed separately for each batch of fuel assemblies transferred to the SFP in the different refueling outages.
3. The assumptions regarding the extent of fuel damaged given an SFP fuel uncovery event occurs are then applied.

What do the results displayed in Figure 8-11 indicate for each offload scenario? The display of the risk measures varies over a wide range. The most optimistic result is for the case labeled “60 days”. This case yields the lowest results for two reasons. One is that fuel overheating is assumed only to occur during the first 60 days after each of the four reactor offloads. After 60 days, the individual fuel assembly heat load is considered too low to overheat even the hottest fuel. Credit is taken for air-cooling by natural convection to limit fuel temperatures. The second reason this case yields low results is that it assumes no propagation of the initially overheated fuel to other fuel assemblies offloaded in prior refueling outages. The risk measures for the four offload scenarios are the same for the case “60 days”. This is because no fuel assemblies that have only recently been offloaded are transferred to the ISFSI.

These same insights also apply to cases labeled “6 months” and “1 year”. Of course, the period of time for which fuel is at risk of overheating increases for these two cases, but the fuel assemblies assessed as at risk of overheating are still restricted to those most recently offloaded from the reactor. Again, there is no difference in the results between the offload scenarios for these two cases because none of the fuel at risk is offloaded to the ISFSI.

The case labeled “All (60 days)” yields higher results than those just discussed and show some risk measure differences between the offload scenarios. Like for case “60 days”, this case assumes that only the most recently offloaded fuel is vulnerable to overheating given a fuel uncovery event and only within the first 60 days. However, if fuel overheats, it further assumes that all other fuel present in the SFP also overheats. This leads to a greater equivalent number of fuel assemblies that overheat. It also reveals differences between the four offload scenarios. This is because now fuel from the many past reactor offloads is assumed affected. Therefore, offload scenarios that remove the most fuel assemblies prior to each reactor offload have the lowest risk measures in this case. The post-shutdown 7-year offload scenario has the highest risk measure since it transfers no fuel assemblies until very late. No fuel assemblies are transferred to the ISFSI until just in time to remove all of them 7 years after

Unit 2 permanent shutdown; i.e., more than 6 years after EOL. As expected, the pre-shutdown vendor option has the lowest results because it transfers the most fuel assemblies at the earliest times. The differences between the fuel offload scenarios are still seen to be small however for this case. It is clear that the contribution to the risk measure is largest when all past reactor offloads of fuel assemblies are assumed to overheat. An important consideration is how long after reactor shutdown such a period should last − 60 days or some other duration.

The cases labeled "1" to "13" show increasing risk measures and greater differences between the offload scenarios as the case number "n" increases. While the Unit 2 reactor is operating, and periodically undergoing refueling, there will always be fuel in the SFP that has been offloaded within the last refueling cycle; i.e., less than approximately 1.5 years since reactor offload. Therefore, there will always be a set of relatively hot fuel assemblies available to first be raised in temperature and potentially overheat other, cooler adjacent fuel assemblies (propagation). The difference in the total results for these cases is attributed to the extent of overheating caused by propagation to older fuel assemblies from previous refueling outages that are still present in the SFP. Our approximate model for campaigns to the ISFSI assumes that the coolest fuel assemblies are transferred first − those that have been in the SFP the longest. This means that if propagation can only involve warmer fuel assemblies that have been offloaded from the reactor in the more recent refueling outages, there should be little difference in the results between offload scenarios. This is what is seen in Figure 8-11 for cases labeled "1", "2", and "3". The differences in the risk measures for these offload scenarios are limited. Only for case "4" and higher do the risk measures separate by offload scenarios, when there are actual differences in the number of fuel assemblies remaining present in the SFP that are assessed as vulnerable to fuel overheating by propagation. The risk measure absolute results and differences between offload scenarios increase further as the value of "n" increases above 4.

The cases named "n" only go up to "13" even though there have been many more refueling outages at Unit 2. A review of the refueling outage records indicates there have already been substantial numbers of fuel assemblies transferred to the ISFSI from the Unit 2 SFP. If only the fuel assemblies cooled the longest were transferred, then none of the fuel assemblies offloaded in the earliest refueling outages would still be in the SFP at the beginning of 2R22 (the start of this assessment beginning May of 2021). With the earliest offload, first transfer to ISFSI assumption, the only earliest offload fuel assemblies that still remain in the SFP at the beginning of 2R22 were offloaded from the reactor in April 2006. There is no need to track earlier refueling offloads since the model assumes all of the fuel from the earlier offloads has already been transferred to the ISFSI.

For case "13", effectively all fuel assemblies in the SFP at the time of the SFP fuel uncovery are assumed to overheat at all points while fuel is in the SFP. Therefore, case "13" represents the bounding case for evaluating the extent of fuel damage given SFP fuel uncovery. The results for case "13" in Figure 8-11 match the values in Table 8-4 and the final time integrated results in Figure 8-6. Table 8-4 and Figure 8-6 did not account for the reduction in the final risk measures by considering the extent of fuel overheating given SFP fuel uncovery.

The reason the pre-shutdown vendor option shows almost no further increase in the risk measure for cases numbered greater than "5" is that by the time Unit 2 is permanently shut down in late August of 2025, there are no more fuel assemblies in the SFP that have been cooling for more than five refueling outages; i.e., none that have been cooling for more than 7.5 years. This is because of the modeling assumption that already transferred them to the ISFSI. By the time of the Unit 2 permanent shutdown, the oldest fuel assemblies still present in the SFP were instead offloaded in September of 2019. Also, by 2R23 in October of 2022, there are no fuel assemblies stored in the SFP offloaded before February, 2018 (only assemblies from the last six reactor offloads are present).

By contrast, the reason the post-shutdown 7-year offload scenario still increases after the Unit 2 permanent shutdown is because all of the fuel assemblies going back to the last 13 reactor offloads are still present in the SFP. The other two offload scenarios (pre-shutdown 7 year and pre- shutdown 5 year) still have fuel assemblies from the prior 11 reactor offloads at the time that Unit 2 reactor is permanently shut down. Therefore, the results for these other two offload scenarios level off for case “11”.

The conclusion from the results of cases “1” to “13” is that there is almost no difference in the risk measures between the offload scenarios up to case “4”. The differences between offload scenarios then increase for case “5” and above. The results for the risk measure consider the extent of fuel that overheats, given fuel uncovery, are very similar for the offload scenarios. Only when it is assumed that fuel assemblies from five or more prior reactor offloads (i.e., that have been cooling for more than 7.5 years) participate in fuel overheating do the differences between offload scenarios become pronounced. This is a significant observation because it requires propagation to fuel assemblies of much lower heat loads before appreciable differences between the offload scenarios are seen.

What are the insights for selecting the assumption set for extent of fuel overheating given fuel uncovery that best characterizes the conditions at DCPP? See Appendix C for summaries of the available literature on this subject. One insight presented in Appendix C is that the extent of fuel propagation is likely a function of fuel assembly heat load. Another insight is that for fuel uncovery events that do not result in Zr-air fires, the extent of fuel overheating may be limited by heat removal away from the SFP that comes without operator intervention (from building leakage, room heat sinks, rejection through the FHB walls and thermal radiation to the outside). Therefore, for DCPP information about the heat load of individual fuel assemblies and for the SFP as a whole is now presented.

Table 8-5 summarizes the heat loads within the DCPP SFP with time after reactor shutdown. The single, average fuel assembly decay heat load with time was computed by taking the core power and dividing by the number of fuel assemblies in the reactor core (193), yielding a power level within the reactor of 17.7 MW per average fuel assembly. The average fuel assembly decay heat was then evaluated assuming 1200 consecutive days in the reactor, with burnups up to 62 GWD/MTU. The decay heat loads with time are provided for a single average fuel assembly, starting at 4 days after reactor shutdown at which time the average fuel assembly decay heat is 66.09 Kw.

The total SFP heat load with time is then presented in Table 8-5 for two situations (for a typical refueling and for the EOL final offload in which the entire reactor core remains offloaded to the SFP) and for each of the four offload scenarios. The single average fuel assembly decay power with time after reactor shutdown (PG&E, 2019j) starts when the fuel assembly is offloaded from the reactor at approximately 4 days after reactor shutdown. These single average fuel assembly heat loads are not specific to any one refueling outage and are provided just for information. The decay heat loads for the times listed are just a subset of the daily heat loads after reactor trip provided by PG&E. The initial heat load for the average fuel assembly is large but is seen to decay quickly with time in the first 6 months. The average fuel assembly power level drops below 1 kW after about 3.5 years of cooling in the SFP.

The full set of average fuel assembly decay heat loads for all times after shutdown was used to estimate the total SFP heat loads that are listed in Table 8-5. Decay times were separately accounted for each prior refueling offload as well as for the typical refueling. These totals were obtained by scaling the average heat load for each fuel assembly modeled as present in the SFP accounting for its decay since it was offloaded from the reactor. For the total SFP heat load for a “typical” refueling outage (the first set of colored headings for all four offload scenarios) the fuel assemblies present for the 2R22 refueling outage, scheduled to begin May 2, 2021, were chosen. The evaluation assumes that 80 fuel assemblies are permanently offloaded for this typical refueling. This refueling outage is selected as representative of the remaining refueling outages. Each of the following refueling outages is expected to be similar until the EOL . Therefore, the heat load entries for this first situation are only presented for the first 1.5 years after reactor offload, the approximate time between refuelings. In subsequent periods between refuelings similar decay heat loads are repeated as in this representative refueling cycle. However, for each of the other refuelings, the exact number of fuel assemblies present is used to evaluate the risk measures.

At DCPP, for each refueling outage the entire reactor core of fuel assemblies (193 in total) are initially transferred to the SFP, until near the end of the refueling outage, when the fresh fuel and fuel assemblies with remaining capacity (113 total) are reloaded into the reactor core. The decrease in total SFP heat load within the first 30 days reflects this transferring of fuel back to the reactor core.

The post-shutdown 7-year offload scenario does not transfer any fuel assemblies from the SFP to the ISFSI until about 6.5 years after the Unit 2 reactor is permanently shut down. The other three offload scenarios do transfer fuel assemblies to the ISFSI, which is why their results are slightly lower. Not shown in Table 8-5 but accounted for in the risk analysis is the changing number of fuel assemblies in the SFP between 4 days and 30 days. The heat generated by the temporary offload of the entire reactor core is accounted for until the time short of 30 days when the new, fresh fuel and the fuel assemblies with remaining capacity are returned to the reactor core.

The last four columns in Table 8-5 present the total SFP heat load (in Kw) with time after reactor shut down for the second situation when Unit 2 is permanently shut down (August of 2025). In this case all the fuel assemblies are offloaded from the reactor and none returned to the reactor. Further, this final offload occurs just 15 months after the previous refueling (2R24). The shorter time between refueling means that fuel assemblies offloaded prior to 2R24 will have had less time to cool than the average fuel assemblies. These differences in cooling time are tracked in the risk evaluation. There are other differences, including that the fresh fuel loaded in 2R24 will have spent less time in the reactor by its end of life than an average fuel assembly. This means that the average fuel assembly decay heat with time may also differ for the end of life offload than is projected in Table 8-5. The differences in fuel assembly residence times in the reactor core are not reflected in this risk evaluation but are not expected to be significant.

The full set of 193 fuel assemblies offloaded at EOL remains in the SFP. The greater number of hot fuel assemblies offloaded accounts for the much higher heat loads throughout the first 1.5 years as compared to the total SFP heat load for a typical refueling outage. The heat load goes to zero between 6.5 and 8.5 years for the EOL situation depending on the offload scenario. The transfer of all the remaining fuel assemblies in the SFP to the ISFSI occurs at different times for the four offload scenarios.

Table 8-5. Summary of Spent Nuclear Fuel Assembly Heat Loads versus Time after Reactor Shutdown

Time After Reactor Shutdown, in Days	Average Fuel Assembly Heat Load (kw)	Total SFP Heat (kw) for Typical Refueling Outage (2R22 Used)				Total SFP Heat (kw) for EOL Permanent Shutdown			
		Post-Shutdown 7-year	Pre-Shutdown 7-year	Pre-Shutdown 5-year	Pre-Shutdown Vendor Option	Post-Shutdown 7-year	Pre-Shutdown 7-year	Pre-Shutdown 5-year	Pre-Shutdown Vendor Option
4 days	66.0857	13,460	13,460	13,460	13,460	13,626	13,502	13,502	13,158
30 days	29.1059	3,021	3,021	3,021	3,021	6,472	6,347	6,347	6,006
60 days	19.9738	2,276	2276	2,276	2,276	4,691	4,567	4,567	4,229
100 days	14.9728	1,858	1858	1,858	1,858	3,703	3579	3,579	3,245
180 days	8.3559	1,300	1232	1,232	1,202	2389	2,266	2,266	1,938
365 (1 year)	4.1828	916	850	850	673	1,522	1,400	1,400	1,085
539 (~1.5 years)	2.6932	766	701	701	328	1,196	1,076	1,076	768
743 (~2 years)	1.8356	NA	NA	NA	NA	1,000	882	882	558
1096 (~3 years)	1.1522	NA	NA	NA	NA	835	719	719	334
1459 (~4 years)	0.8673	NA	NA	NA	NA	758	645	645	170
1824 (~5 years)	0.7448	NA	NA	NA	NA	717	607	0	91
2191 (6 years)	0.6869	NA	NA	NA	NA	691	583	0	33
2555 (~7 years)	0.6550	NA	NA	NA	NA	0	0	0	7.0
2958 (~8 years)	0.6331	NA	NA	NA	NA	0	0	0	0.0

The extent of fuel overheating can also be different for different fuel uncovery sequences. For this assessment, fuel uncovery events involving leakage from the SFP are considered to potentially result in Zr-air fires. Fuel uncovery events without leakage are judged to not result in Zr-air fires. Table 8-6 groups the fuel uncovery sequence contributors to total SFP fuel uncovery frequency by sequence type. The sequence types are those judged to have similar characteristics for extent of fuel damage and cesium release, given that SFP fuel uncovery occurs. Only sequences where the SFP coolant levels fall below the coolant baseplates at the bottom of the fuel assemblies by coolant leakage are judged to potentially lead to sufficient air natural circulation to result in a Zr cladding fire (see EPRI,2014, Section F.2.4). The flow rates from the natural circulation must be high enough to enable self-sustaining cladding oxidation and yet low enough to not cool the cladding sufficiently to limit their cladding temperatures below the 900°C limit where self-sustaining oxidation would occur. From Table 8-6 it is found that 92.3% of the fuel uncovery sequence frequency for DCPP does not involve coolant leakage and so is judged to not result in Zr-cladding fires. Only 7.7% involves small or larger size leakages and could potentially result in such cladding fires. Our evaluation of whether fuel uncovery sequences would result in cladding fires depends on the time of the fuel cycle when the seismic event occurs, the corresponding fuel assembly decay heat available, and our assumptions about how these factors are related.

Table 8-6. Fuel Uncovery Sequence Groupings for Assessment of Extent of Fuel Damage Given Fuel Uncovery

% of total Fuel Uncovery Frequency	Sequence Categories	Event Tree Sequences Included	Model for Extent of Fuel Damage
66.1%	No SFP Leakage, SFP Cooling Fails	4, 5, 7 (include LERF additions from 6 and 13)	Fuel Collapse and Blockage Before Coolant Drops Below Bottom of Active Fuel. Zr Cladding Fires Unlikely
26.2%	Auxiliary or FHB Collapse, No SFP Leakage Sequences	16	Fuel Collapse and Blockage Before Coolant Drops Below Bottom of Active Fuel. Zr Cladding Fires Unlikely
5.8%	Small SFP leakage Sequences	10,11,12,14	Fuel Collapse and Blockage Before Coolant Drops Below Bottom of Active Fuel. Zr Cladding Fires Unlikely
1.9%	Moderate SFP Leakages	15 (includes frequency where both auxiliary building and SFP fail)	Coolant Level Drops Below Fuel Assembly Baseplates Allowing Natural Circulation of Air Through Fuel Assemblies

The modeling assumptions used to determine the extent of fuel overheating and hence the amount of cesium released for the two groupings of sequences listed above are described below. These assumptions are for the case where FHB ventilation has not been enhanced by opening doors at the lower and upper building elevations. The seismic sequences contributing the most to seismic initiated fuel uncovery are those in which the seismic accelerations are so large that these actions are not credited even hours after the initial shaking has occurred. The fuel handling building atop the auxiliary building has several large area doors on opposing sides of the building that can promote natural air flow within the structure. The simple metal siding could also be modified to increase airflow over the spent fuel pools if the doors were not functional. The procedural guidance for these actions is described in Section 7. Such actions are not credited in the following assessment.

Table 8-7 presents general criteria for the extent of fuel overheating and hence the release of cesium for different SFP conditions. These criteria were adopted for the development of a base case assumption set for this study. Both sequences with short times to fuel uncovery and long times to fuel uncovery are covered by these modeling criteria. Only sequences with short times to fuel uncovery (e.g., leakage events from the bottom of the SFP) are considered for possible Zr-air self- sustaining oxidation, and therefore the potential for propagation from hotter fuel assemblies to other fuel assemblies. Table 8-8 interprets the criteria listed in Table 8-7 as they apply to sequences with long times to fuel uncovery (losses of SFP cooling with no SFP leakage). Table 8-9 interprets the criteria listed in Table 8-7 for sequences involving short times to fuel uncovery.

Table 8-7. Criteria for Overheating of Spent Fuel under Different Conditions of Fuel Assembly Heat Load and Coolant Inventory

Criterion ID	General Criteria for Extent of Fuel Overheating Given Fuel Uncovery	Comments
1	When the most recently offloaded fuel assemblies have heat loads greater than 15 kw each, it is expected that all the fuel assemblies in the SFP will overheat. An average fuel assembly is below 15 kw after a cooling time of 100 days.	This criterion is applicable for leakage or no leakage fuel uncovery sequences. The choice of this criterion was largely prompted by the integrated fuel assembly heat up tests (SANDIA, 2015). 1.8 MW is the total SFP heat load for a typical refueling offload for DCPP after 100 days cooling. For an EOL full core offload, a total SFP heat load of 1.8 MW is reached after about 10 months of cooling, or as soon as 8 months of cooling if there are pre-shutdown fuel assembly transfers to the ISFSI.
2	When the average fuel assembly has a power level less than 15 kw but greater than 8 kw and the SFP loses its water inventory in a short time, Zr-air ignition occurs, there is propagation to other fuel assemblies, and the most recent two full cores of fuel assemblies overheat; i.e., five reactor offloads of about 80 fuel assemblies each. For DCPP this amount of fuel assembly propagation corresponds to the most recent five reactor offloads of fuel being susceptible. Fuel assemblies offloaded from the reactor earlier than these five offloads are not susceptible to overheating.	For the first 6 months of cooling in the SFP, the DCPP average fuel assembly heat load is greater than 8 kw. The integrated fuel assembly heat up tests for PWR fuel assemblies (Sandia, 2015 and USNRC, 2016a) revealed no Zr-air ignition for fuel assemblies with less than 8 kw, implying it is still possible somewhere between 8 kw and 15 kw. So self-sustaining Zr-air oxidation is postulated to occur for fuel assemblies in this range of heat loads but the extent of propagation is reduced. The stated criterion is applicable even without forced flow ventilation or natural convection in the FHB via a "chimney" effect; i.e., building leakage is sufficient. (USNRC, 1979, NUREG/CR-0649). The average fuel assembly in the most recent five reactor offloads affected by propagation each has heat loads greater than 0.64 kw. After 6 months cooling, the total SFP heat load for a standard refueling outage is 1.2 MW, and for an EOL full core offload it is about 2.3 MW.

3	After 6 months of cooling when an average fuel assembly heat load is less than 8 kw, but the fuel has cooled for less than 3 years, the hotter than average fuel assemblies may still reach the self-sustaining Zr- air oxidation temperatures − at 3 years the average fuel assembly heat load is 1.15 kw). This is possible if the <u>SFP loses its water inventory in a short time</u>, and there is no FHB ventilation. Zr-air ignition occurs, there is propagation to nearby fuel assemblies, and the most recently offloaded two full cores of fuel assemblies from the reactor at Unit 2 overheat. For DCPP this amount of propagation corresponds to the most recent five reactor offloads of fuel being susceptible. Fuel assemblies cooled for more than 3 years are judged not susceptible to Zr-air self-sustaining oxidation even without FHB ventilation.	This criterion is applicable even without forced flow ventilation or natural convection of the FHB via a "chimney" effect; i.e., building leakage is assumed sufficient. A very well-ventilated FHB can reduce the 3 years cooling time to as little as 280 days (0.77 years of cooling), if the average burnup is 33 GWD/MTU (USNRC,1979, NUREG/CR-0649). No credit is taken for the FHB being well ventilated in this assessment because it would require local operator actions to initiate the chimney effect after a very large seismic acceleration event (above 3.0 g SA). This criterion is more restricted than might be inferred from the fuel assembly heatup tests (SANDIA, 2015) in that fuel assemblies with heat loads less than 8 kw are nevertheless counted as susceptible to Zr-air self-sustaining oxidation. This conservative criterion is judged to allow for the possibility that some fuel assemblies with larger power peaking factors may still be susceptible. At DCPP, for the first 3 years of cooling in the SFP the average single fuel assembly heat load is greater than 1.15 kw. The average fuel assembly in the most recent five reactor offloads each have a heat load greater than 0.66 kw. During a standard refueling cycle of 1.5 years, the total SFP heat load is greater than 0.77 MW unless there are many pre-shutdown transfers of fuel assemblies to the ISFSI (as in the vendor option). For an EOL full core offload, for the first 3 years of cooling, the total SFP heat load is greater than 0.8 MW, except for the three offload scenarios that include pre-shutdown transfers to the ISFSI.

4	After more than 3 years decay, Zr-air oxidation ignition is assumed not to occur; i.e., for a single average fuel assembly heat load less than 1.15 kw, regardless of the total SFP heat load or the sequence leading to fuel uncovery. Some higher heat load fuel assemblies may still overheat as described in criterion 5.	MELCOR runs for BWR fuel assemblies (EPRI, 2014, Page E-5) indicate that air-cooling, for sequences where the SFP loses its water in a short time, would limit cladding temperatures to less than 527°C provided the single fuel assembly heat loads are less than 3 kW. The same study indicates that fuel assemblies with heat loads greater than 4.5 kw would not be coolable. The amount of FHB ventilation assumed for these calculations is not described. For an SFP with higher burnup PWR fuel assemblies and with ventilation limited to building leakage, a more conservative criterion is chosen; i.e., more than 3 years of cooling and individual fuel assemblies with less than 1.15 kw of heat load. Fuel overheating may still occur without sustained Zr-air oxidation taking place. See the next criterion below.
5	Fuel overheating occurs for the fuel assemblies whose average heat load is greater than 0.7 kw, but only if the total SFP heat load is greater than 0.5 MW. A minimum total SFP heat load is introduced, below which natural convection processes involving FHB leakage should be able to remove the entire SFP heat load without overheating the fuel assemblies with relatively low heat loads (less than 0.7 kw).	No calculations of the amount of heat levels that may be removed by FHB building leakage are available. Additional FHB openings created by seismic failures, or by operator actions to open FHB doors would certainly increase this heat removal limit. Future calculations would have to consider not only the potential for higher natural convection flows, but also heat transfer from the fuel assemblies themselves to areas above the SFP. An average fuel assembly heat load of less than 0.7 kw is reached after 6 years of cooling in the SFP; i.e., roughly the most recent four reactor offloads. For a standard refueling outage, the most recently offloaded average fuel assembly heat load does not drop below 0.7 kw before the next refueling outage occurs. Also, the total SFP heat load does not drop below 0.5 MW before the next refueling outage, except for the vendor option and only then for the last portion of the refueling cycle after many cooler fuel assemblies have been transferred to the ISFSI. For the EOL full core offload, the total SFP heat load drops below 0.5 MW just before the SFP is emptied for three of the offload scenarios. For the fourth, pre-shutdown vendor option, the total SFP heat load drops below 0.5 MW between 2 and 2.5 years of SFP cooling.

Table 8-8. Conditions for Extent of Fuel Overheating Given a Long Time to Fuel Uncovery Occurs; i.e., No Coolant Leakage.

Times after Reactor Shutdown	Standard Refueling - Partial Core Offload	EOL Offload - Full Core Offload
4 to 100 days	Between 4 and 100 days of cooling, average fuel assemblies have heat loads greater than 15 kw and so all fuel assemblies present overheat. The total SFP heat load is above 1.8 MW for less than 100 days of cooling	Average fuel assemblies have heat loads greater than 15 kw and so all are assumed to overheat. The SFP total heat load is above 3.3 MW for the entire period.
100 days to ~10 months	Fuel overheating occurs only for the fuel assemblies whose average heat load is greater than 0.7 kw and only if the total SFP heat load is greater than 0.5 MW. Only fuel which has been transferred to the SFP in the last ~ 6 years (four reactor offloads) exceed the 0.7 kw criterion. These two heat load conditions are true for this entire period for three of the offload scenarios.	Fuel overheating occurs only for the fuel assemblies whose average heat load is greater than 0.7 kw and only if the total SFP heat load is greater than 0.5 MW. Only fuel which has been transferred to the SFP in the last four reactor offloads exceeds the 0.7 kw criterion. Total SFP heat load drops below 1.8 MW (about 10 months). The duration of cooling until the total heat load is less than 1.8 MW varies depending on the SFP offload scenario. For total SFP heat loads less than 1.8 MW within the first 10 months, the criterion for the next time interval applies.

~10 Months until next refueling (~1.5 years)	Fuel overheating occurs only for the fuel assemblies whose average heat load is greater than 0.7 kw and only if the total SFP heat load is greater than 0.5 MW. Only fuel which has been transferred to the SFP in the most recent four reactor offloads exceed the 0.7 kw criterion. For three offload scenarios the total SFP 0.5 MW condition is met for the entire period. For the vendor option offload scenario, the total SFP heat load drops below 0.5 MW just before 1.5 years.	Fuel overheating occurs only for the fuel assemblies whose average heat load is greater than 0.7 kw and only if the total SFP heat load is greater than 0.5 MW. Only fuel which has been transferred to the SFP in the most recent four reactor offloads exceed the 0.7 kw criterion. The total SFP 0.5 MW condition is exceeded for the entire period.

~1.5 Years until SFP Empty	This time interval is not applicable for a standard refueling since fuel assemblies from the next reactor offload restart the applicable conditions. Note: the 1.5 years duration is an approximation. The time between reactor offloads varies between refueling outages.	Fuel overheating occurs only for the fuel assemblies whose average heat load is greater than 0.7 kw and only if the total SFP heat load is greater than 0.5 MW. The average individual fuel assembly heat load is greater than 0.7 kw for approximately 6 years. Total SFP heat load drops below 0.5 MW for the Post-Shutdown 7-year and Pre-Shutdown 7-year offload scenarios just prior to emptying the SFP of all fuel assemblies at 7 years. Pre-Shutdown 5-year off load scenario drops below 0.5 MW before 5.5 years. The Pre-Shutdown vendor option offload scenario drops the heat load below 0.5 MW after about 2.5 years.

Table 8-9. Conditions for Extent of Fuel Overheating Given a Short Time to Fuel Uncovery Occurs due to Coolant Leakage

Times after Reactor Shutdown	Standard Refueling - Partial Core Offloads	EOL Offload - Full Core Offload
4 to 100 days	Between 4 and 100 days of cooling, average fuel assemblies have heat loads greater than 15 kw and so all fuel assemblies present overheat. The total SFP heat load is above 1.8 MW for less than 100 days of cooling.	Average fuel assemblies have heat loads greater than 15 kw and so all are assumed to overheat. The SFP total heat load is above 3.3 MW for the entire period.
100 days to 6 months	Zr-air ignition occurs for average fuel assemblies with greater than 8 kw heat load (i.e., less than 6 months of cooling) and due to propagation, the most recently offloaded two full cores of fuel assemblies overheat. For DCPP this amount of propagation corresponds to the most recent five reactor offloads of fuel being susceptible. The susceptible fuel assemblies may have cooling times up to 7.5 years, and have heat loads greater than 0.64 kw each. Fuel assemblies offloaded from the reactor earlier than these five offloads are not susceptible to overheating. SFP total heat remains above 1.2 MW during the first 6 months cooling following a partial refueling.	Zr-air ignition occurs for average fuel assemblies with greater than 8 kw power (i.e., less than 6 month of cooling) and the most recently offloaded two full cores of fuel assemblies overheat. For DCPP this amount of propagation corresponds to the most recent five reactor offloads of fuel being susceptible. The susceptible fuel assemblies may have cooling times up to7.5 years; i.e., and have heat loads greater than 0.64 kw each. Fuel assemblies offloaded from the reactor earlier than these five offloads are not susceptible to overheating. SFP total heat remains above ~2.0 MW during the first 6 months of cooling.

6 months to 3 years	Between 1.5 years and 3 years this criterion is not applicable since the next refueling offload of hot fuel assemblies restarts the applicable conditions. Zr-air ignition occurs for fuel assemblies with less than 1.5 years of cooling (i.e., with heat loads greater than 2.65 kw) and propagate to the most recent five reactor offloads of fuel assemblies. The fuel assemblies in these five reactor offloads all have heat loads greater than 0.64 kw. SFP total heat load varies depending on the offload scenario and decay time; at 1.5 years it ranges from 0.3 to 0.77 MW depending on the specific offload scenario evaluated.	Zr-air ignition occurs for fuel assemblies with less than 3 years of cooling (i.e., with heat loads greater than 1.15 kw) and propagate to the most recent five reactor offloads of fuel assemblies. The fuel assemblies in these five reactor offloads all have heat loads greater than 0.64 kw. Total SFP heat load varies depending on the offload scenario and decay time; at 3 years it ranges from 0.3 to ~0.8 MW depending on the specific offload scenario evaluated.

Greater than 3 years until the SFP is empty	Not applicable since next refueling offload of hot fuel assemblies restarts the applicable conditions	Zr-air ignition for fuel assemblies with more than 3 years of cooling (i.e., with heat loads greater than 1.15 kw power) does not occur. Fuel overheating may still occur but only for the fuel assemblies which have cooled for less than 6 years (i.e., whose average heat load is greater than 0.7 kw), and only if the total SFP heat load is greater than 0.5 MW. A 6-year cooling time is roughly the last four reactor offloads of fuel assemblies. Post-Shutdown 7-year and Pre-Shutdown 7-year offload scenarios have total SFP heat loads greater than 0.5 MW almost until the SFP is emptied seven years after EOL. Pre-Shutdown 5-year and Pre-Shutdown vendor option scenarios offload enough assemblies to the ISFSI to drop below 0.5 MW within 5 years and 2.5 years of EOL respectively.

The criteria in Tables 8-8 and 8-9 for the two types of fuel uncovery sequences were then used to develop a base case assessment for the extent of fuel overheating and hence the amount of cesium released from fuel. These features were then combined with the first two aspects of the risk measure (i.e., probability of fuel uncovery in a time interval, and the equivalent number of fuel assemblies present at the time of the fuel uncovery) to evaluate the risk measure results for all four offload scenarios. Two sets of results were generated; one for each of the two types of fuel uncovery sequences which were weighted by their respective occurrences frequencies to obtain the final risk measures.

The risk measure results are presented in Table 8-10. The gray column repeats the results from Table 8-4 for the risk scores which did not consider the extent of fuel overheating; i.e., the integrated probabilities of SFP fuel uncovery weighted by the number of fuel assemblies present adjusted for cesium decay. Recall that the risk scores conservatively assume all fuel assemblies present in the SFP released all their cesium, assuming all fuel overheated with probability 1.0. The first two columns headed in blue are results for the two types of fuel uncovery sequences which are then weighted by frequency in the next blue column to obtain the base case results. The third blue column gives the base case risk measure including consideration of the extent of fuel overheating, given a fuel uncovery event. The base case risk measures can be directly compared to the gray column which assumed all fuel overheated, given a fuel uncovery event. As expected, the results are lower for the base case having accounted for the fact that not all fuel present in the SFP may overheat. The final column in Table 8-10 compares the results for the four offload scenarios as a percentage of the post- shutdown 7-year offload scenario. The pre-shutdown vendor option is seen to have roughly half the risk of the other three offload scenarios.

Figure 8-12 compares the contributions to the base case risk measures cumulated over the assessment period (while there is SNF in the SFP) for the four offload scenarios. These results can be compared with the time-dependent results in Figure 8-6 which conservatively assumed that all fuel present in the SFP overheats given a fuel uncovery event. The extreme right of these time-dependent cumulative curves matches the base case risk measures in Table 8-5. Figure 8-12 shows that the pre- shutdown vendor option risk measure separates from the other three offload scenarios before the Unit 2 EOL full core shut down and remains below the other three offload scenarios throughout. The pre-shutdown vendor option stops increasing after 2028 when the SFP total heat load drops below 0.5 MW (refer to the base case criteria for extent of fuel overheating documented in Table 8- 7).

Table 8-10. Comparison of Fuel Uncovery Probabilities Weighted by Extent of Fuel Overheating and Cesium for Offload Scenarios

Offload Scenario (Years refers to SFP empty time after Unit 2 EOL)	Seismic SFP Total Probability of Fuel Uncovery Until Empty, weighted by effective # of FAs in SFP, No Fuel Overheating	No Leakage Uncovery Sequences	Leakage Uncovery Sequences	Base Case - Integrated Probability of Fuel Uncovery Weight by Extent of Fuel Overheating	Ratio of Weighted Total Fuel Overheat Probabilities to Post-SD 7-year offload scenario (%)
POST-SD 7-year	0.139	0.067	0.074	0.067	100.0%
PRE-SD 7-year	0.119	0.065	0.072	0.065	97.1%
PRE-SD 5-year	0.099	0.055	0.062	0.056	82.8%
PRE-SD Vendor Option	0.053	0.035	0.046	0.036	52.8%

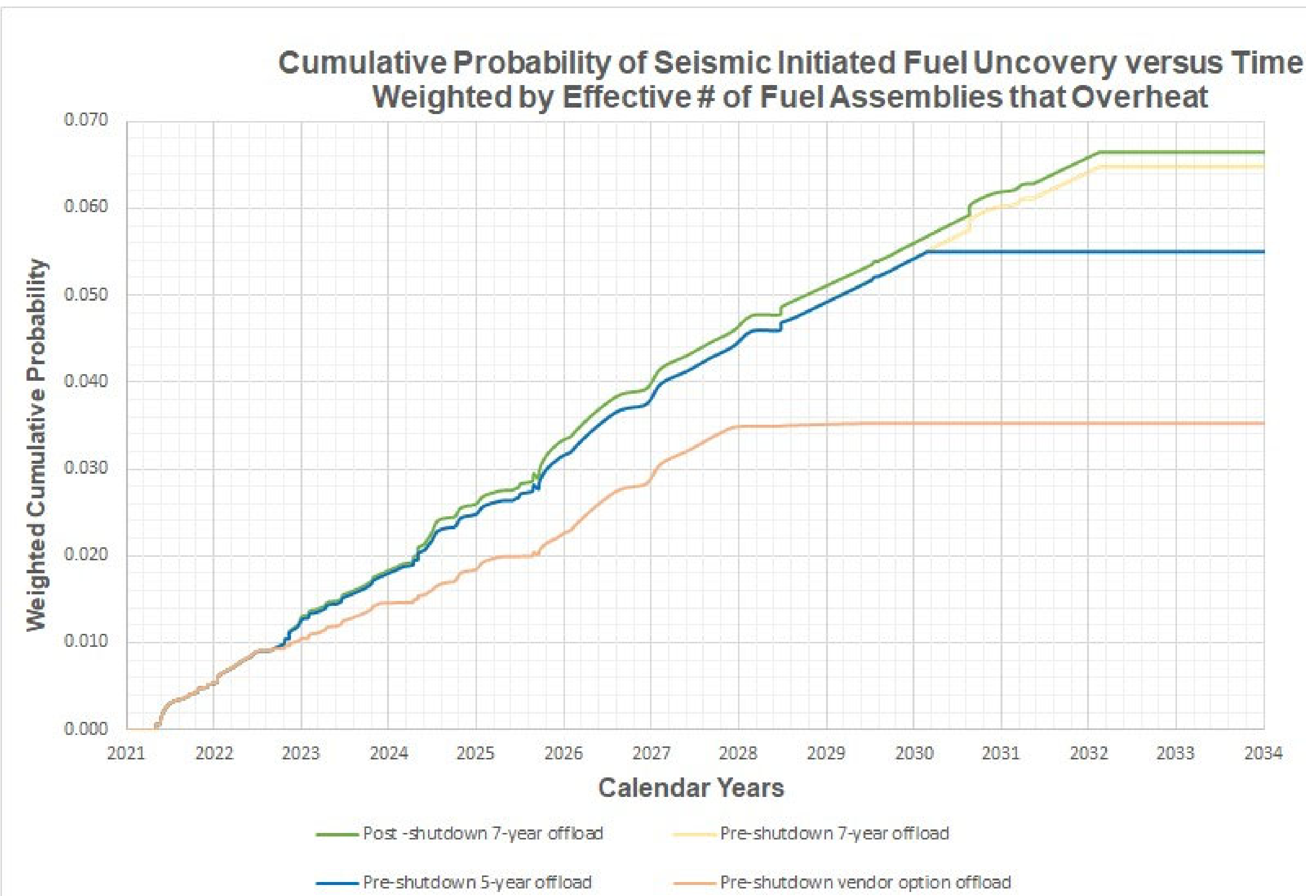


Figure 8-12. Cumulative Probability of Seismic Initiated Fuel Uncovery versus Time Weighted by the Number of Fuel Assemblies Overheated in the SFP and Adjusted for Cesium Decay

The risk measures for each offload scenario presented in Table 8-10 apply only to the risks of storage and some handling of transfer casks within the FHB. Given the results in Table 4-7, it was argued in Section 4 that the risks posed by transferring the SNF from the FHB to the ISFSI and the risks from SNF storage at the ISFSI contributed minimally. One question that may be asked is whether there is less risk to the public in total by moving SNF earlier from the SFP to the ISFSI, considering the increased risks that would result from the added SNF at the ISFSI.

The decrease in risk from the SFP caused by transferring 32 MPCs loaded with SNF to the ISFSI 2 years earlier than would otherwise be the case, can be obtained by just comparing the risk measures in Table 8-10 between the pre-shutdown 7-year offload scenario and the pre-shutdown 5-year offload scenario. Both of their campaigns to remove the remaining fuel assemblies occur after the EOL but are completed at different times thereafter. The decrease in risk measures between the pre-shutdown, 7-year and the pre-shutdown, 5- year offload scenarios is (.0654-.0557) = 9.7E-3. The decrease in risk measures for the SFP would presumably differ even more if the fuel assemblies are transferred prior to the EOL.

For purposes of comparing the SFP and ISFSI risks, a bounding assessment is made of the added risk caused by transferring 32 MPCs to the ISFSI two years early. This bounding assessment is to evaluate the added risk at the ISFSI caused by adding the 32 MPCs for two years prior to when they would otherwise have been added. The same initiating events assessed as applicable to the ISFSI listed in Table 4-5, and their frequencies, are applied for two years to these added 32 MPCs. Each initiating events' frequency is increased in proportion to the number of MPCs then stored at the ISFSI (i.e., 57) as compared to the storage overpacks present immediately before the final campaign of 32 MPCs. This added frequency, for each initiating event, is judged to represent an upper bound on the added frequency caused by the greater foot print of storage overpacks present at the ISFSI. It is conservatively also assumed that all storage overpacks affected by each initiating event affect only the MPCs transferred two years early. The risk to all other storage overpacks transferred prior to the final campaign remains the same as presented in Table 4-5. After weighing each added initiating event frequency applicable to the 32 MPCs transferred early, the increase in the risk measure at the ISFSI is computed to be just 8.32E-5. This represents only 0.86% of the decrease in the risk measures for the SFP (i.e., 9.7E-3) between pre-shutdown, 7-year and the pre-shutdown, 5-year offload scenarios. The conclusion of this exercise is that transferring MPCs early from the SFP to the ISFSI decreases the total risk measure for SNF handling and storage at DCPP. This conclusion is reached though likely overstating the added risk at the ISFSI.

The uncertainties in the extent of fuel overheating on the final risk measures have been examined earlier in this subsection. The results of that initial assessment are displayed in Figure 8-11. One additional variable for that assessment is the assumption of natural heat removal capability in limiting the extent of fuel overheating after extended periods of cooling while stored in the SFP. Specifically, Table 8-7's criterion 5, defined for the base case, was that fuel overheats in the long term if the individual fuel assembly heat load is greater than 0.7 kw, and the total SFP heat load is greater than 0.5 MW. The 0.5 MW value was assumed as a reasonable amount of heat removal just based on building leakage, room heat sinks, and rejection through the walls and thermal radiation to the outside in the absence of an enhanced natural convection condition that the operators might implement by opening doors of the FHB. The 0.5 MW amount of heat removal is less than what can be readily be removed by forced flow using the FHB ventilation fans, and much less than what can be removed by an enhanced natural convection alignment of the FHB doors. Table 8-11 considers this uncertainty further via a sensitivity analysis.

The sensitivity results presented in Table 8-11 indicates the change in the extent of fuel overheating, given variations in the assumed natural heat removal capability. For a typical refueling offload of a partial reactor core, the total SFP heat load drops below 1 MW in just less than 1 year. For a full core offload such as after end of life, the SFP total heat load drops below 1 MW after a longer period but still just less than 1.5 years. Therefore, for the sequences without coolant leakage in which Zr-air fires are not expected to occur, the 1 MW sensitivity case just involves setting all fuel overheating to zero for times when the total SFP heat load drops below 1 MW.

For fuel uncovery sequences with coolant leakage, Zr-air self-sustaining oxidations are modeled as occurring up to 3 years after the reactor offload, regardless of the amount of heat removal and to propagate the overheating to the most recent 5 reactor offloads of fuel assemblies. Therefore, for fuel uncovery sequences with leakage, the extent of fuel overheating with cooling less than 3 years after reactor offload is unchanged. Only after more than 3 years of cooling following the end of life full core offload, are the evaluated amounts of fuel overheated changed to zero. The 3 years cooling time is governing for the leakage sequences with fuel uncovery after the full core offload.

The sensitivities of the risk measures for the four offload scenarios are presented in Table 8-11. The results are also shown pictorially in Figure 8-13. As expected, the smaller the heat removal assumed, the larger the risk measures. For each sensitivity case, the post-shutdown 7-year offload scenario always yields the highest risk measure and the pre-shutdown vendor option always yields the lowest risk measure. Coincidentally, the base case assumption of 0.5 MW for the heat removal capability yields the largest difference in the risk measures between these two enveloping offload scenarios. The differences between the risk measures for these two offload scenarios get smaller as the assumed heat removal capacity increases.

It is noted that the results between the pre-shutdown 7-year and pre-shutdown 5-year offload scenarios do not differ when the assumed natural convection heat removal is increased to greater than 0.5 MW. In these cases, the number of fuel assemblies present are the same during each of the three refuelings and for the EOL time interval, the fuel assemblies are also the same until the time that the final campaign is scheduled. The final campaign doesn't occur until the reduced heat load in the SFP is less than 0.75 MW. Therefore, the risk measure results of these two offload scenarios are the same.

Table 8-11. Sensitivity of Integrated Fuel Uncovery Probabilities Weighted by the Extent of Fuel Overheating for Different Amounts of Natural Convection Heat Removal

Sensitivity Cases for Extent of Fuel Overheating	POST-SD 7-year	PRE-SD 7-year	PRE-SD 5-year	PRE-SD Vendor Option
Base Case with 0.0 MW Cutoff	0.0691	0.0672	0.0575	0.0485
Base Case with 0.25 MW Cutoff	0.0686	0.0667	0.0571	0.0439
Base Case 0.5 MW Cutoff	0.0673	0.0654	0.0557	0.0356
Base Case with 0.75 MW Cutoff	0.0558	0.0455	0.0455	0.0256
Base Case with 1.0 MW cutoff	0.0394	0.0327	0.0327	0.0194
Base Case with 1.5 MW cutoff	0.0266	0.0226	0.0226	0.0150
Base Case with 2.0 MW cutoff	0.0226	0.0206	0.0206	0.0129

Recall that the extent of fuel overheating, given a fuel uncovery event, is evaluated as the sum of the contribution from SFP fuel uncovery leakage sequences and SFP fuel uncovery no leakage sequences. For the base case risk measures, the leakage sequences contribute about 8% to the total; largely this is due to the greater frequency of no leakage sequences that still lead to fuel uncovery. As the heat removal capacity is assumed to increase, not only do the risk measures decrease, but also the relative contribution of the leakage sequences increases to as much as 24% of the total for the pre-shutdown vendor option when the heat capacity is assumed to be 2 MW. This is because the modeled contribution from Zr-air self-sustaining oxidation is not being affected by the assumed natural convection heat removal capacity.

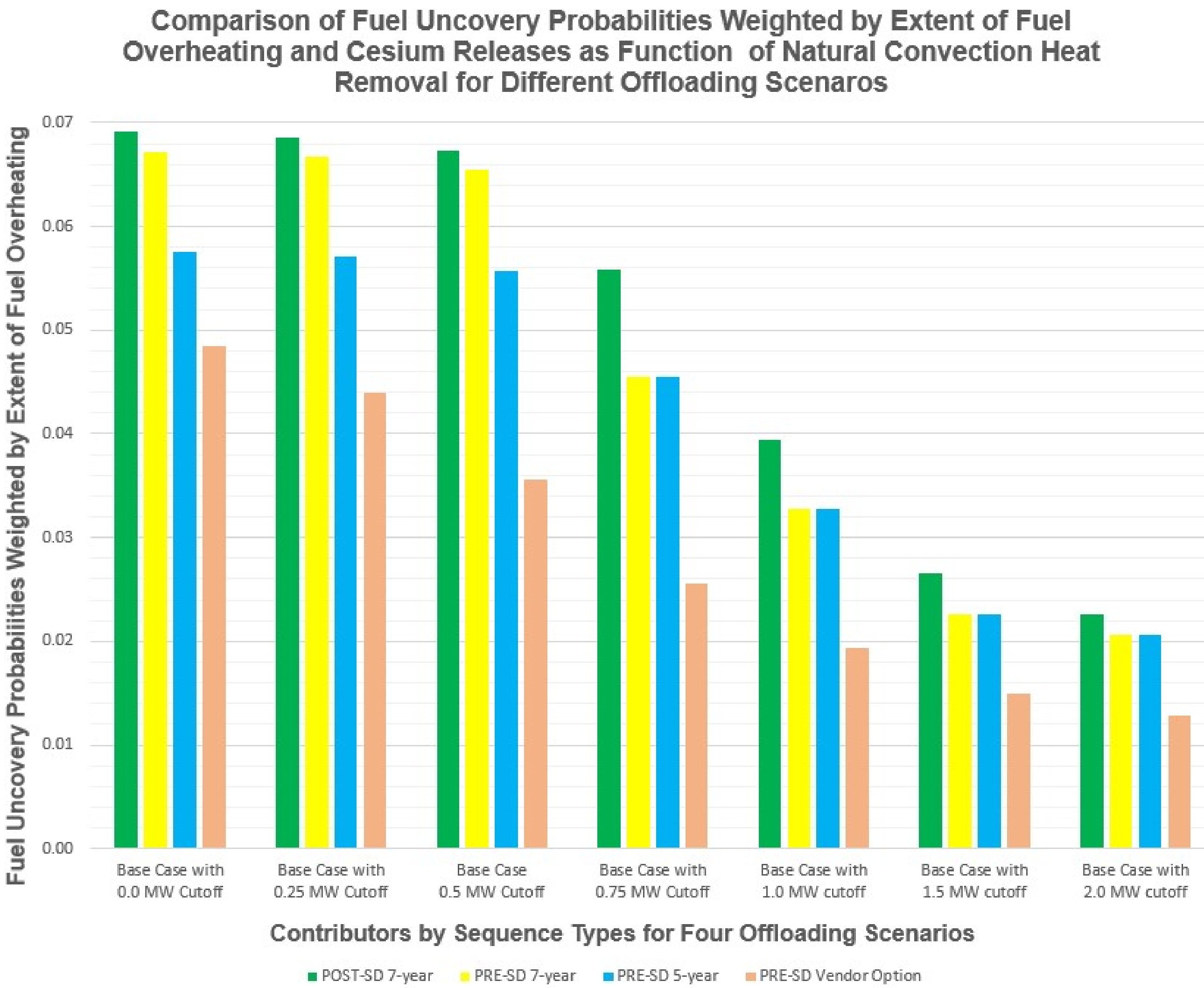


Figure 8-13. Sensitivity of Probability of Seismic Initiated SFP Fuel Overheating Versus Total Amount of Natural Convection Heat Removal Expressed in MW

The evaluated risk measures are a convenient way to aggregate the accident sequence frequency and consequence results into one number for each of the four offload scenarios. This approach is useful for decision making, but makes the important simplification that risk as represented by the full set of risk triplets can be represented as the sum of the products of individual sequence frequencies and their consequences. This is not always the case as some decision makers choose to emphasize the importance of accident sequences with larger releases of cesium in a more risk adverse way than just proportional to the equivalent number of fuel assemblies overheating.

Figure 8-14 displays an alternate approach to summarizing the risk results in a way that more fully displays the risk results, providing a picture of the role of larger consequence fuel uncovery accident sequences.

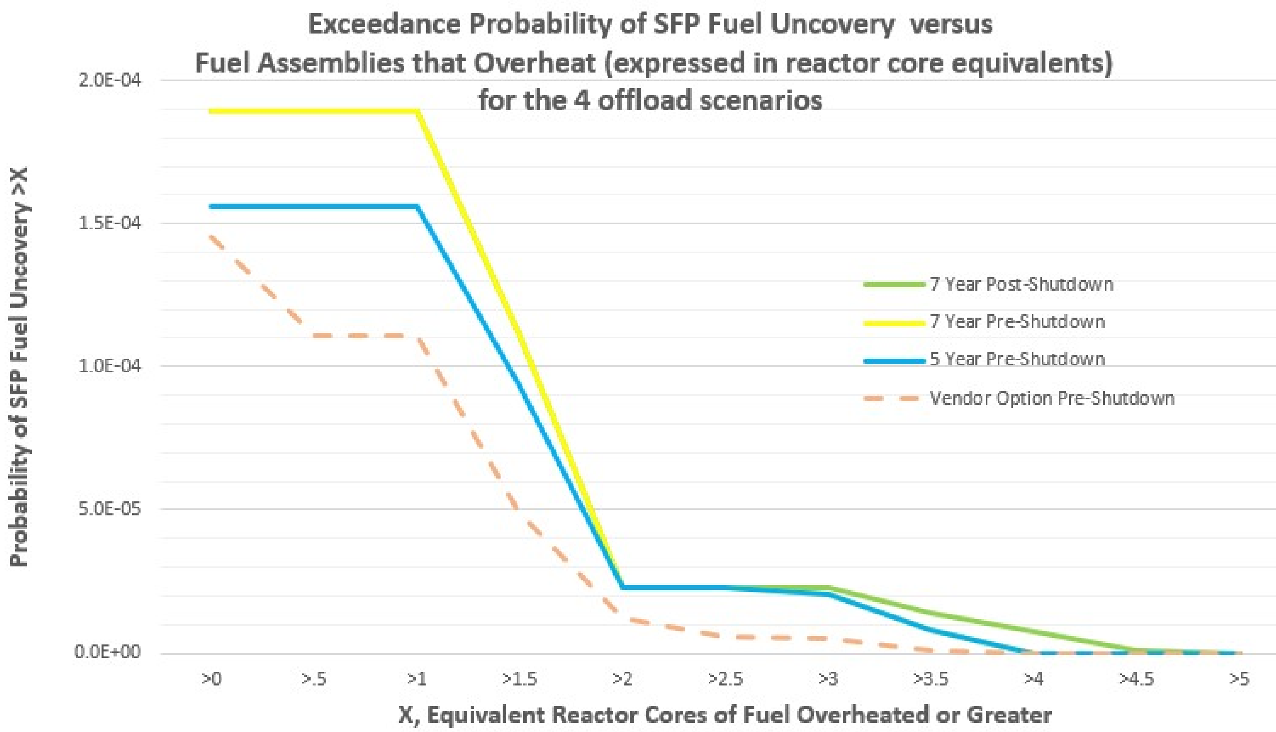


Figure 8-14. Probability of Seismic Initiated SFP Fuel Uncovery while Fuel Assemblies are in the SFP versus Number of Equivalent Reactor Cores of Fuel Assemblies that Overheat

Figure 8-14 shows the exceedance probability curves for each for the four offload scenarios. Here the exceedance probability is the integrated fuel uncovery probability restricted to the accident sequences which involve a selected amount of cesium released due to fuel assemblies overheating or greater. For a given value of X, if sufficient fuel assemblies are not present in the SFP at the time of the fuel uncovery, or there are enough present, but not that many fuel assemblies overheat, then the risk contribution from that time interval are not included in the probability sum for that value of X. Since the Y-axis displays exceedance probabilities the exceedance probability for at least two equivalent cores of fuel or greater overheat is always less than the exceedance probability of one core equivalent. For convenience, the equivalent number of fuel assemblies overheating is expressed in multiples of reactor cores. At DCPP each reactor core holds 193 fuel assemblies.

The vertical axis of Figure 8-14 is in units of fuel uncovery probability, summed over the entire time that SNF is present in the SFP and if a fuel uncovery occurred would lead to some fuel overheating. The vertical axis is not weighted by the number of fuel assemblies that would overheat. Therefore, the maximum point on the vertical axis is less than the seismic total SFP fuel uncovery frequency (1.74E-5 per year) multiplied by the number of years with SNF in the SFP for each offload scenario.

The number of fuel assemblies overheating portion of the risk measure is captured by the horizontal axis. The fuel uncovery probability plotted, only includes the time intervals in which a given number of fuel assemblies, X or more, would overheat releasing their cesium inventory. By this display, the integrated fuel uncovery probability for releasing more cesium than the equivalent of 1.5 reactor cores for the vendor option offload scenario is just 5E-5. The integrated fuel uncovery probability for releasing more cesium than the equivalent of at least one reactor core, again for the vendor option, is about 1.25E-4, and is the lowest of the 4 offload scenarios evaluated. The maximum value on the horizontal axis is less than the peak number of fuel assemblies in the SFP for all offload scenarios because not all fuel assemblies present are always susceptible to overheating.

Figure 8-14 illustrates that the vendor option offload scenario leads to lower exceedance probabilities of occurrence in the remaining time that spent fuel will be present in the SFP. This is true for all ranges of equivalent number of reactor cores worth of fuel assemblies that are assumed to overheat. For example, if the amount of overheated fuel considered is one reactor core equivalent or greater, the vendor option has an exceedance probability of occurrence of 1.25E-4, the 5-year pre-shutdown offload scenario is at 1.7E-4, and the post-shutdown and pre-shutdown 7-year offload scenarios are both at 2.0E-4. Figure 8-14 also illustrates that for events involving 2 or more reactor core equivalents, the probabilities do not differ much between the four offload scenarios—all have occurrence probabilities less than 3.0E-5. Note that the Y-axis values are dimensionless exceedance probabilities. They should not be confused with units of frequencies per year. For Figure 8-14, the durations in years that the fuel assemblies are present in the SFP is already considered in summing the applicable time interval probabilities of fuel overheating. A conclusion from this figure is that while the vendor option is slightly favored, these differences are more limited for SFP fuel uncovery events that result in larger amounts (greater than 2 reactor cores) of fuel overheating.

8.5 Analysis Uncertainties and Study Assumptions

The quantitative risk analyses documented in the preceding sections uses mean, or point estimate values for the model parameters. Use of mean values are preferred for model parameters as they also reflect the influence of the uncertainties Example parameters are for seismic hazard frequencies and for seismic capacities for both structures and components. For our discussion here, model input parameters are divided into two classes according to their impacts on the model results; i.e., parameters that affect the risk measures for all offload scenarios equally, and risk parameters whose values affect the risk measures of the offload scenarios differently.

First is discussed the model parameters that impacts the risk results for all four offload scenarios nearly the same. These parameters affect the frequency of seismic initiated SFP fuel uncovery which is then applied equally to all four offload scenarios.

The uncertainties in this class of parameters may call into question the conclusion reached in Section 3, that all four offload scenarios are already safe. For seismic risk assessments, the parameters with the largest uncertainty in this class are the seismic hazard frequencies. The uncertainties in the seismic hazard frequencies typically dominate the overall uncertainty in a seismic risk assessment. Uncertainties in the plant structures' and components' seismic capacities also may contribute but are generally much less an influence.

The seismic hazard exceedance family of curves were assessed as part of the long term seismic program (LTSP) for the DCPP site and have been refined over many years. Figure 8-15 presents the current hazard curve family (PG&E, 2020). By exceedance frequency, this means the sum of all seismic event frequencies with the acceleration listed on the x-axis, or greater. The curves labeled by percentiles in Figure 8-15 are identified by the degree of belief that the true exceedance frequency for a given acceleration is less than or equal to the value on the y-axis. So, at 3.0 g SA, the upper most curve shows that there is a 95% confidence that the exceedance frequency is less than about 7E-5 per year. The range of uncertainties between the different seismic hazard exceedance curves varies with the acceleration of the seismic event chosen. Figure 8-15 clearly illustrates how the uncertainties get larger for the higher acceleration seismic events.

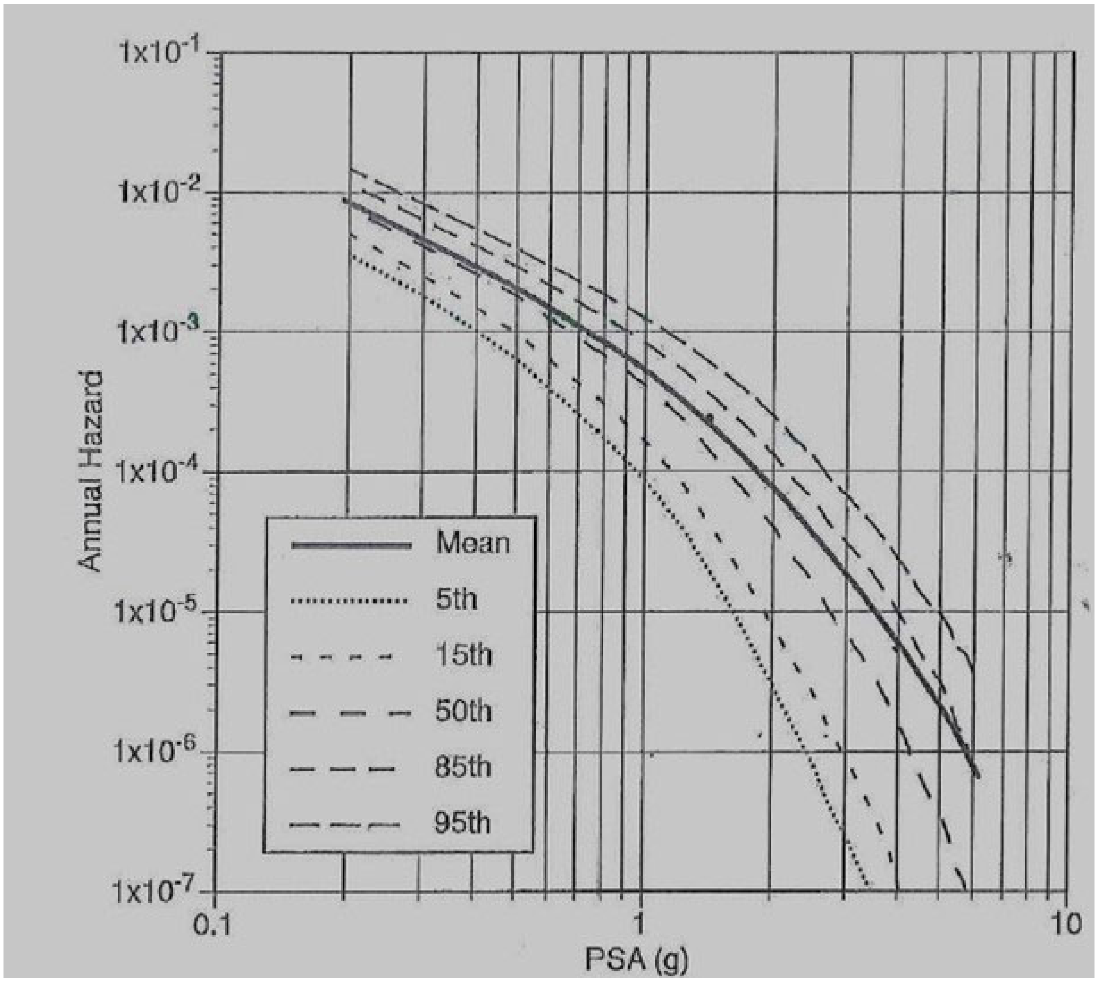


Figure 8-15. Seismic Hazard Curve Family for the DCPP Site – 5Hz ground motion

One question is whether the uncertainty in the seismic hazard exceedance frequency curves is large enough to void the conclusion in Section 3 (that the risk of SFP operation at DCPP is already safe) when a more limiting seismic hazard curve is chosen as input to the model for SFP fuel uncovery. It is convenient for this discussion to examine the uncertainty in the seismic hazard exceedance frequency for one specific acceleration. An acceleration of 3.0 g SA is chosen as representative since the largest contribution to seismic initiated SFP fuel uncovery frequency comes from that acceleration or higher. The exceedance frequencies for selected percentiles of the seismic hazard curve family at 3.0 g SA, in events per year for the DCPP site are presented below.

5% = 2.57E-7
50% = 7.02E-6
Mean (or ~73%) = 1.75E-5
95% = 6.80E-5
100% = 1.90E-4

The "range factor" statistical measure is typically used as a measure of the uncertainty. It is computed using the above values: range factor = $(95\%/5\%)^{1/2}$ = 16.3. The range factor is a measure of the spread of the uncertainty distribution both above and below the median, or 50% value. The base case analysis in this assessment uses the mean hazard curve for quantification. The mean exceedance frequency of 1.75E-5 events per year at 3.0 g SA has the same exceedance frequency as does the 73% of the hazard curve family. Use of the exceedance frequency at the 95%, which has a higher frequency, instead of the exceedance frequency from the mean curve would therefore increase the estimated seismic initiated SFP fuel uncovery frequency. Use of the 95% curve would therefore increase the SFP fuel uncovery frequency by the ratio; (95%/mean) = 3.89; i.e., by about a factor of 4. The upper 95% percentile curve is typically used as a high estimate when investigating the range of exceedance frequencies for seismic events. If instead the 100% seismic hazard exceedance frequency is chosen as a very high estimate, then the increase in exceedance frequency would be; (100%/mean) = 10.9.

A factor of 4 multiplier on the base case SFP fuel uncovery frequency of 1.74E-5 per year (from Table 4-7) would then yield a high estimate for the SFP fuel uncovery frequency of 7.0E-5 events per year. This high estimate for the DCPP Unit 2 SFP fuel uncovery frequency is a factor of 2 higher than that assessed by the USNRC for the set of plants assigned to Group 4 PWR with twin SFPs (3.46E-5 per year, as described in Section 3). Using the high estimate for seismic hazard exceedance frequency from the 95% curve, the risk from seismic initiated SFP fuel uncovery would still meet the QHOs with plenty of margin. The scaled latent cancer risk from the DCPP SFP fuel uncovery would still only be 1.5% of the QHO for latent cancer risk. The public health consequences would still result in zero prompt fatalities since the amount of cesium released in any fuel uncovery event is unchanged by this sensitivity. Use of the high estimate from the seismic hazard exceedance 95% maintains the conclusion that SNF operation at DCPP would still be considered safe by comparison to the USNRC's QHOs. Even using the very high exceedance frequency estimate from the 100% hazard curve, the latent cancer risk from seismic initiated SFP fuel uncovery for DCPP would still only be 4.2% of the USNRCs QHO for latent cancer risk.

The conclusion concerning the first class of parameter uncertainties which affect all offload scenarios equally is that even using very high estimates for the seismic hazard exceedance curve the risks from DCPP SFP operation would still meet the USNRC's QHOs by a large margin.

A second parameter of this first class are the human error probabilities used for recovery. The sensitivity results presented in Figure 8-10 showed that lowering the human error probabilities for recovery from a fuel uncovery condition at seismic accelerations above 3.0 g SAS would markedly decrease the seismic caused SFP fuel uncovery frequency. In the base case model, the human error probabilities for recovery from an SFP fuel uncovery event are each set to 1.0 for accelerations above 3.0 g SA. Therefore, there are no higher human error probabilities which would change the base conclusion that SNF operation at DCPP is already safe.

The assessment of human error probabilities affecting the transport of SNF from the FHB to the ISFSI has not been performed. It is plausible that transferring the loaded MPCs in fewer campaigns each with more MPCs transferred sequentially could reduce the potential for human errors. Using the same operational teams in larger campaigns may lead to fewer errors overall. Fewer operational errors may reduce the potential for crane drops or misplacements during the fuel transfer processes. It may be difficult to ensure use of the same operational teams for more frequent campaigns spread out over several years. The pre-shutdown vendor option offload scenario specifies a larger number of smaller campaigns spread out over a longer duration than is planned for the other three offload scenarios (Table 8-2). An assessment of such organizational factors was not considered in this study. Higher probabilities for crane drop or misplacements would have to increase substantially compared to the industry failure rates to be comparable to the risks from seismically caused SFP fuel uncovery events. See the comparisons by locations in Table 4-7 and item 11 in Appendix B for more discussion on crane drop frequency.

The modeling assumption whose Class 2 parameter uncertainty is judged to potentially impact the offload scenario rankings is Item 17 in Appendix B. This key assumption covers a range of modeling parameters that are used to evaluate the amount of radiological release from an SFP that undergoes a fuel uncovery event. This assumption also considers the veracity of using a surrogate risk measure rather than extending the risk analysis to quantify the potential for offsite public health risk. The risk measure approach is judged appropriate because a similar intermediate result was used in the USNRC's assessment of SFP operating risk.

This study's objective is to investigate the differences in SNF storage and handling risks for the different offload scenarios. Meeting this objective requires an assessment of the amount of cesium release given an SFP fuel uncovery event occurs. This assessment of cesium release varies for each time-period while the SFP contains fuel assemblies. There is not an accepted, integrated thermal-hydraulics software tool that directly models the release of cesium caused by an SFP fuel uncovery event. The modeling assumptions needed to compute the risk measure for each offload scenario are described in Section 6 and earlier subsections of this section.

Figure 8-11 presents the results of sensitivity cases evaluated to determine the change in the risk measures for each offload scenario given different assumptions determining the extent of fuel overheating and by inference the amount of cesium released. The conclusion from that set of sensitivity cases was that the older the fuel assemblies are cooled and still remain vulnerable to overheating, the greater the difference in the computed offload scenario risk measures. In the limiting sensitivity case (Case 13 in Figure 8-11), the amount of cesium released is evaluated as 100% from all SNF present. In this most conservative Case 13 the offload scenario risk measures vary (Table 8-10) from 0.139 for the post-shutdown 7-year offload scenario to 0.053 for the pre-shutdown vendor option offload scenario, a reduction of about 62%. The other two offload scenarios have risk measures between these two.

The base case risk measures evaluated using a more realistic set of assumptions (Tables 8-7, 8-8 and 8-9) vary from 0.0673 for the post-shutdown 7-year offload scenario to 0.0356 for the pre-shutdown vendor option offload scenario (see Table 8-10), a reduction of about 47%.

Table 8-11 presents the sensitivity of the risk measures for each offload scenario to one Class 2 parameter that impacts the extent of cesium release given an SFP fuel uncovery event occurs. The parameter is the assumed amount of heat removal from the SFP by natural convection. The natural convection referred to is due to FHB leakage, without any action to open FHB doors or fans. So, such actions which would certainly increase the amount of heat removal and limit the extent of fuel overheating were not credited in this risk assessment. The base case heat removal assumed is 0.5 MW. Varying the assumed amount of heat removal by natural convection from zero to 2.0 MW does not change the risk ranked order of the offload scenarios but does narrow the difference in risk measures between them. For the base case, the risk measure in Table 8-10 for the pre-shutdown vendor option was 47% lower than for the post-shutdown 7-year offload scenario. Taking no credit for natural convection (i.e., 0 MW), both risk measures are higher than in the base case and the vendor option is reduced by just 30% compared to the post-shutdown 7-year offload scenario. At the other extreme for natural convection heat removal (2 MW) both risk measures are lower, and the vendor option is reduced 43% compared to the post-shutdown 7-year offload scenario.

It is concluded from this discussion that the uncertainties in the extent of fuel overheating given an SFP fuel uncovery event are not so large as to alter the risk ranking of the offload scenarios. Their values are estimated to change plus or minus by a factor of 1.5 depending on the assumptions. While the rank order does not change the absolute risk measures can change significantly. This is apparent from Figure 8-11 and Table 8-11. The key point is that the chosen assumption set changes the risk measures for all four offload scenarios similarly. The risk measure differences between offload scenarios vary only moderately. Even considering an extreme assumption set, the predicted DCPP SFP operation public health risks compare favorably with other U.S. studies.

9 Conclusions

This study has assessed the radiological risks associated with storage and movement of Diablo Canyon Power Plant (DCPP) spent nuclear fuel from the spent fuel pools (SFP) to the Independent Spent Fuel Storage Installation (ISFSI). The study began by reviewing the SNF handling and operating activities which could pose a risk to public safety. A comprehensive list of potential initiating events was developed from the literature and knowledge of DCPP SNF operations. The initiating events include heavy load drops, transit accidents, equipment failures, extremely severe seismic events including slope sliding, loss of offsite power, human failures and other external events such as aircraft crashes and explosions, all as a function of the different activities involved. These initiating events were then screened for applicability and contribution to the risks of DCPP SNF handling and storage. The frequencies of the retained initiating events were then evaluated and the potential for radiological releases on the affected fuel assemblies from each initiating event at each location were assessed. Three DCPP locations containing SNF were considered: 1) within the SFP located at the FHB, 2) the SNF as it is transported in transfer casks from the FHB to the ISFSI, and 3) the storage of SNF at the ISFSI. By this approach, it was concluded that the risk to the public was dominated by extreme seismic events and cask drops at the SFP. These initiating events could result in an extended period of fuel uncovery and overheating of some or all of the fuel assemblies present.

This conclusion of key initiating event contributors is consistent with other nuclear industry studies. Earlier studies by the USNRC (USNRC, 2013) of these same initiating events concluded further that public health risks would be well below the USNRC's quantitative health objectives (QHOs) for prompt fatalities and for latent cancer risk. Their analyses postulated bounding amounts of radioactivity release for any SFP fuel uncovery event, and the bounding amount was applied for all times. No credit was taken for offloading any SNF from the SFP. Their analyses also simulated the transport and dispersion of the radiation release offsite and evaluated its potential impact to members of the public. The frequency of such release events and the amounts of radiation that could realistically be released from the SFPs at DCPP have been assessed as lower than was postulated by the USNRC for its generic PWR plant. The existing SNF operating practices for DCPP therefore should have risks that are sufficiently low as to meet the QHOs by wide margins.

Therefore, this assessment focused on the risks caused by seismic and cask drop events during SNF handling and storage of fuel assemblies located within the SFP for comparison with the USNRC's results. This assessment also considered the time-dependent amounts of fuel assemblies present within the SFP. Thereby, the relative risks of the four offload scenarios integrated over time with fuel assemblies in the SFP could be compared. This time-dependent assessment considers the fact that the date the SFP is emptied varies with each offload scenario.

The assessment considered four alternative offload scenarios. The conclusions are provided as responses to the questions posed in Sections 1 and 8 as the basis for this assessment. The questions and responses follow.

1. *Which offload scenario is the least vulnerable to a beyond design basis radiological event, that is, which scenario of the four considered is most likely to assure public safety? How do the four offload scenarios rank?*

The ranking of the four offload scenarios is based on a risk measure defined explicitly for the handling and storage of *DCPP* spent nuclear fuel within the SFP. In particular, the measure is defined as "*the probability of fuel uncovery weighted by time (and duration) and by the equivalent number of* *fuel assemblies which overheat and release cesium (adjusted for cesium decay), given a severe event including beyond design basis events."* This risk measure accounts for the following risk characteristics.

- Frequency of spent fuel uncovery within the SFP
- Duration and time dependent number of fuel assemblies in the SFP
- Extent of overheating of the fuel assemblies present, given fuel uncovery
- Decay of cesium within each fuel assembly since it was offloaded from the reactor

The risk ranking of the offload scenarios is as follows. The numbers in the parentheses from Table 8-10 are based on the risk measure defined above.

- Pre-Shutdown Vendor Offload (0.036)
- Pre-Shutdown 5-Year Offload (0.056)
- Pre-Shutdown 7-Year Offload (0.065)
- Post Shutdown 7-Year Offload (0.067)

The pre-shutdown vendor option offload has the lowest risk measure. Its risk measure is, however, less than 50% lower than the risk measures of the other three offload scenarios; i.e., the risk measures do not differ substantially.

2. *How does the difference in risk between the SFP and the ISFSI impact the safest operating strategy?*

The SNF is less of a threat to the public when in dry storage at the ISFSI, as compared to wet storage in the SFP. The risks to the public by transfer of SNF from the FHB to the ISFSI is small compared to the risk at both these storage locations. The risk of dry storage at the ISFSI is only 7.5% (Table 8-3) as much risk when comparing the frequencies for potential fuel overheating at the ISFSI with the frequencies of SFP fuel uncovery. The DCPP total frequency of SFP fuel uncovery of 1.74E-5 per year is noticeably higher by factors of 3 to 10 than the SFP fuel uncovery frequencies developed in previous studies for other sites (see Table 4-2). The higher seismic initiator contribution for DCPP is a result of the much higher seismic hazard. The more seismically robust design for the DCPP SFP liner and structures limits the seismic SFP fuel uncovery frequency increase.

A more direct comparison of the risks of the two storage locations has been made. This comparison is between the risk reduction from the SFP and the risk increase at the ISFSI when transferring a number of MPCs loaded with fuel assemblies 2 years earlier than planned (See Section 8.4). It is found that even a bounding assessment of the increase in risk measures at the ISFSI is less than 1% of the decrease in risk measure for the SFP. The offload scenario risk measure rankings show that transferring fuel assemblies earlier to the ISFSI decreases the risk at the SFP, and this decrease is only minimally offset by the increase in the risk measure at the ISFSI. As noted in the response to Question 1, the differences in risk between the four offload scenarios considered are modest. There are also heat load limits on the MPCs used to transfer SNF to dry storage which limit the amounts and earliest times that fuel assemblies can be transferred to the ISFSI.

3. *What additional technology is needed to perform a full scope probabilistic risk assessment of the SFP and the ISFSI with the same confidence as those typical of contemporary nuclear power plant PRAs, particularly with regard to the quantification of the uncertainties?*

In this assessment of risks at the SFP and ISFSI, the following technical limitations were encountered.

For the SFP:

a. A realistic development of the seismic capacities is required of the SFP structure and liner, including considerations of their likely failure modes and detailed knowledge of the liner openings and potential drainage paths through the concrete structure. At present the HCLPF acceleration for the seismic capacity curve has been evaluated but not the full capacity curve.
b. An approach needs to be developed for evaluating the potential for air-cooling by natural convection within the SFP and fuel handling building to determine the extent of fuel overheating and for assessing the resulting radiological releases from uncovered fuel to the environment, i.e., an SFP- specific MELCOR (Sandia, 2017, Sandia, 2017a) type modeling capability is needed.
c. An assessment of the potential for fuel overheating and the extent of radiological releases, given fuel uncovery is needed.
d. The absence of severe accident sequence-specific calculations of the onsite radiation environment precluded knowing the effect they would have on the ability to perform the directed mitigating operator actions. A modeling assumption was made to limit such credit for sequences judged to involve high radiation releases from the reactor core. Recovery actions to respond to events involving only the SFP should occur before any radiation releases from the fuel assemblies within the SFP.
e. A detailed assessment is needed of the potential impacts of seismically initiated severe accidents involving the reactor during power operation. The Fukushima accident revealed the potential for hydrogen releases from the reactor containment into nearby plant locations containing SFP cooling equipment.

For the ISFSI:

a) A realistic evaluation of storage overpack and concrete pad seismic capacities is needed including for accelerations well beyond the design basis. An assessment of seismic capacity to the point of failure or to very low seismic frequency is desired. Failure modes which could lead to the overheating of stored fuel assemblies are of interest.

b) Extension of the thermal analysis performed to date involving the storage overpack is required to account for the decreases in radiological inventories and decay heat levels for different times after reactor offload. The licensing calculations used in this risk assessment consider only the most bounding heat loads permitted within a storage overpack.

Technical limitations for both the SFP and ISFSI:

a) Many more load lifts have been conducted throughout the industry since the original heavy load drop frequency per movement was assessed. This additional evidence could be included in a reassessment of the heavy load drop frequency.

b) DCPP site specific modeling of time-dependent radiological releases from either storage location, their subsequent transport by winds offsite, and the potential effects on public health, considering evacuation would be helpful. At present bounding release assumptions used in the USNRC's site modeling for a generic PWR plant are used for risk comparisons against the USNRC's Quantitative Health Objectives.

4. *How does the current spent fuel risk analysis compare the spent fuel risk with the DCPP risk of reactor core damage?*

As noted in Table 9-1, the mean frequency of events initiating fuel uncovery in the SFP is just 22% of the nuclear plant total reactor core damage frequency. The DCPP-specific seismic hazard curves (PG&E, 2018b) are used in both these frequency assessments. The SFP fuel uncovery frequency is still less than the seismic only contribution to the reactor core damage frequency.

The seismic initiated mean recurrence interval for SFP fuel uncovery (~57,000 years) is about five times the DCPP Unit 2 total CDF mean recurrence interval of 12,000 years. That is, the frequency is lower for the SFP fuel uncovery than for the reactor CDF. The frequency is higher and so the recurrence interval for SFP fuel uncovery is less than half that for the Unit 2 reactor large early release events (127,000 years). In contrast to this assessment of SFP fuel uncovery risks, only about one-third of the reactor CDF comes from seismic events and two-thirds of the reactor LERF. The comparison with the severe reactor accident LERF in Table 9-1 is for information only.

The comparison of SFP risk and nuclear plant risk requires an assessment of whether the SFP fuel uncovery releases would be large. Whether all fuel assemblies contribute to the radiologic release during an SFP fuel uncovery depends on the effectiveness of air-cooling by natural convection and potential SFP spray effectiveness in removing radionuclides from the effluent stream. As noted earlier, this is an area requiring further study.

Table 9-1. Severe Accident Frequencies and Recurrence Intervals for DCPP Unit 2

Severe Accident Frequencies and Recurrence Intervals for DCPP Unit 2	Frequency (events per year)	Recurrence Interval (years)
Unit 2 Seismic SFP Fuel Uncovery	1.74E-05	57,000
Unit 2 Total Reactor CDF	8.15E-05	12,000
Unit 2 Total Reactor LERF	7.90E-06	127,000
Seismic Only Unit 2 Reactor CDF	2.83E-05	35,000
Seismic Only Unit 2 Reactor LERF	5.17E-06	193,000

5. *What are the most important variables affecting SFP risk?*

Analyses indicate that the most important variables to SFP risk are the human error probabilities assigned for high acceleration seismic events. The operator actions accounting for the human error probabilities have to do with the proper alignment of alternative paths for SFP cooling and for SFP coolant makeup. These actions are credited when the structures and equipment related to the SFP for these functions remain operational. A key uncertainty for such sequences is the capability for air-cooling by natural circulation within the SFP that could limit fuel overheating, potentially precluding a radiological release. The conditions when air-cooling by natural convection would be successful, without operator actions, can greatly influence the total SFP risk for each of the four proposed offload scenarios.

6. *How do the uncertainties in the assessment impact the results?*

There are several sources of uncertainty in this assessment. The uncertainties introduced by the assumptions listed in Appendix B are examples of sources of uncertainty. Five important sources of uncertainty are highlighted below and discussed in Section 8.5.

- Seismic hazard frequencies
- Seismic capacity of key structures
- Human error probabilities
- Heavy load drop frequency
- Extent of fuel overheating, given an SFP fuel uncovery event

The spent fuel uncovery frequency is driven by seismic events, whose uncertainty is largely due to the uncertainties in the seismic hazard curves. The uncertainties in the seismic hazard curve are large and vary with the acceleration range. Typically, the uncertainties are largest for the extremely high accelerations. In this regard, the frequency of SFP fuel uncovery from seismic events does not differ for the four offload scenarios and therefore it does not affect their rank-order. There has been considerable effort in characterizing the seismic hazard and further analysis is not expected to change the ranking of offload scenarios.

The capacity of the key structures such as the fuel handling and auxiliary building can determine the impact of seismic events on the level of severity of the accident— severity levels having reference to whether it involves a loss of cooling, a leak, or a structural collapse. Structural collapses are likely to preclude access for recovery actions. These types of impacts affect the frequency of extended spent fuel pool uncovery. The impact of the cause of the fuel uncovery on radiological release is a secondary effect, thus a secondary effect on the ranking of the offload scenarios.

Uncertainties in human error probabilities dictate the recovery from the seismic impacts and therefore the spent fuel pool uncovery frequency. Credit for local operator actions is difficult to justify for large seismic accelerations. Again, while the uncertainties may be high, it is believed to have only a secondary effect on offload scenario ranking. The uncertainties in heavy load drop frequencies are largely judgmental and are not fully based on event records. The range of uncertainty is estimated as a factor of ten lower and a factor of ten higher than that assumed. The contribution from drops is less than one in a thousand contribution to the SFP uncovery frequency. While there are statistical data on cranes that are not single failure approved, the data is yet to be fully processed for single failure approved cranes. This assessment indicates the heavy load drops leading to spent fuel uncovery are only a very small fraction of the total spent fuel pool uncovery frequency. Drops that occur at the cask transfer facility may potentially result in mechanical damage to fuel assemblies, but have been analyzed to determine they would not result in fuel overheating. Frequencies of drops at the CTF are also very low. Thus, this uncertainty is not expected to affect the ranking of offload scenarios.

The uncertainties associated with overheating of fuel, given a fuel uncovery event, are significant and do impact the ranking of the offload scenarios. The extent of fuel overheating, given fuel uncovery, can be characterized by two attributes. First is the number of prior reactor offloads of fuel vulnerable to overheating and second, the capability of natural processes for heat removal in the absence of being able to provide cooling. Limited overheating of the fuel would minimize the differences in the offload scenarios. On the other hand, substantial overheating of most of the fuel assemblies present would increase the differences in the offload scenario risk measure rankings. If more of the fuel present would overheat given a fuel uncovery event, this would add to the preference of the vendor option from a risk perspective. However, changing the assumption about the extent of fuel overheating would not change the fact that the risks from each of the four offload scenarios would still be well below the USNRC's quantitative health objectives criteria.

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Appendix A

Reviews of Potential Initiating Events Within and Outside of the Fuel Handling Building

Table A-1. Potential Initiating Events for Fuel Handling and Storage inside the Fuel Handling Building

Initiating Event Category	Analysis of relative risk	Frequency Events/ Year	Potential for Release
A. Heavy load and fuel assembly drops without external influences			
Lift the transfer cask containing an empty multi-purpose canister (MPC) above the low-profile transporter (LPT) and move it into the Unit 2 CWA;	Moving the empty transfer cask (transfer cask) into the Unit 2 cask washdown area (CWA) constitutes a negligible risk: no spent nuclear fuel (SNF) is involved in the move, the transfer cask is attached to the LPT, and even if the LPT with the transfer cask tipped over, as would be more likely during a seismic event, damage would not result in a radionuclide release.	NA	
Move the transfer cask + empty MPC to the SFP Unit 2: move through the hot machine shop and into the FHB/AB bay area of Unit 2 and position adjacent to the SFP Unit 2	The transfer cask containing the empty MPC is lifted 12 inches off the floor using the single failure-proof fuel handling building crane during this activity. Since the lifting height is limited and no SNF is involved, the consequence of a cask drop is negligible. The consequence would still be negligible if a seismic event occurred during the cask lift.	NA	
A.1.b - Position the transfer cask/MPC over the cask recess area of the SFP and lower into the SFP platform structure. The transfer cask restraint provides seismic restraint while the transfer cask is on the platform.	This activity involves a high, heavy lift over the edge of the SFP and into the SFP. The USNRC (USNRC, 2012) cites six crane hoist/drop issues at nuclear plants, although this is an incomplete list. Human reliability analysis (HRA) factors include, among others, inadequate crane/hook inspection and maintenance, improper rigging and rigging failure. A single failure proof crane has a drop probability of 9.6E-8/ movement (USNRC, 2001, App 2C). From the same reference, accounting for the critical path areas over the SFP and the conditional probability of leakage from the spent fuel pool (SFP) given the drop (assuming 1.0 for the drops from the maximum height) a frequency of 2E-9/ movement is obtained. This rate for loss of coolant due to the fall is further reduced here by a factor of 10 (USNRC, 2014) to account for the DCPP specific analysis performed by PG&E (PG&E, 2019f) showing no coolant leakage would occur. For 80 lifts total for both units this translates to 1.6E-8 loss of coolant probability from drops in 10 years, or 1.6E-9/ year. The USNRC (USNRC, 2007a) cites an ANSI standard (ANS, 1983) that provides standards regarding use of single failure-proof cranes and use of stops, interlocks, or design to prevent the cask crane from passing over the spent fuel. The USNRC (USNRC, 2012), Section 3.2 states: "Other than potential breach of a cask, the primary consequence of concern for a heavy load drop in these plant types is damage to the spent fuel pools (e.g., penetration of the pool)." The *potential* consequences involving radioactive release *outside the nuclear power plant (NPP)* are variable with zero release being by far the most probable. NUREG-1738 (USNRC, 2001) summarized the relative risks for cask drops determined from the two studies: NUREG-1864 states (USNRC, 2007) "Cask drops are considered to pose the greatest potential risks to the public, and fuel misloading events are considered to be much less of a concern, but neither type of event poses a risk anywhere near as high as other risks typically encountered during NPP operation." Section 9.1.2.3.11, Regulatory Position 5.b, notes (PG&E, 2018) that the USNRC determined that: "PG&E has evaluated the drop of a loaded transfer cask from the highest point in the lift to the bottom of the cask recess area in the SFP. The postulated drop consists of 4.67 feet in air followed by 42.83 feet in water. Analysis demonstrates the adequacy of the affected structures during the postulated drop, showing that the drop will not cause (1) loss of building structural function; (2) damage to the SFP resulting in loss of SFP water; or (3) unacceptable damage to other systems or equipment. The SFP stainless steel liner may be damaged in this drop; however, the structural integrity of the concrete forming the SFP is maintained, preventing any uncontrolled leakage." Nevertheless, this analysis pessimistically assumes there is a non-zero, albeit small (i.e., 0.1), probability of a leak from the SFP due to a cask drop of sufficient magnitude that the SFP water level cannot be maintained.	1.60E-09	All fuel assemblies present in SFP uncovery per year

Initiating Event Category	Analysis of relative risk	Frequency Events/ Year	Potential for Release
Load the intact assemblies and damaged fuel cans (if any) in the pre-planned positions in the MPC	The operations during this activity are similar to those described above. The possible scenarios leading to a breach of either the fuel assembly being moved or to other assemblies in the storage racks would be the assembly being transferred hitting other equipment in the SFP or dropping of the assembly over the storage racks or elsewhere. The minimum number of individual fuel assembly moves to load 80 storage overpack casks with an MPC capacity of 32 assemblies would be 80*32=2560. Assuming the individual assembly drop probability is 9.6E-8 per movement (USNRC, 2001, App 2C), then this translates into a 2.5E-5 probability of dropping one assembly. However, in NUREG- 1864 (USNRC, 2007), Section 3.3.1 states the probability of an assembly drop per lift is 3.2E-5. This translates to a total drop frequency of 8.21E-2. For the purpose of this study, the larger value was adopted. However, damage would be limited to the assembly being moved or potentially to one or two additional assemblies in the storage racks. Based on (USNRC, 2007), the probability of failure of fuel cladding is zero for a drop of up to 20 feet. Therefore, it is likely not failed in this proposed event. Even if it was, the damage should be limited to cause 100% of the rods in that assembly to rupture, thereby releasing 30% of the fission gas, 2E-4 of the volatiles, 3E-5 of the mass fraction release as fuel fines and 100% of the crud that spalls off the cladding (USNRC, not dated, Interim Staff Guidance). Even adding this increased probability of assembly damage and the consequence of a partial release of radionuclides from that assembly, compared to potential consequences from scenarios involving a cask drop into the SFP followed by pool drainage, the consequence from partial release of radionuclides from a single assembly would be small. (PG&E, 2016), Section 9.1.2.3.8, states that release of any radioactivity from a damaged fuel assembly from the fuel handling building would be mitigated by: (1) the FHB is held under slightly negative pressure; (2) the plant technical specifications prescribe operability requirements for equipment relied on for assembly movement; and (3) post-accident iodine removal via high-efficiency particulate air (HEPA) filters; and use of charcoal filters upon higher radiation levels in the FHB exhaust air. The assessed release to the environment, at most, is therefore relatively small. The probability of dropping a fuel assembly or a fuel assembly being damaged by contacting other structures or equipment in the SFP would be higher if a seismic event occurred during the assembly move - as would the probability that the FHB functions would partially fail. No analyses on the increased probability of assembly drop or mitigation caused by a seismic event is available however. The estimated low level of radioactivity release is used to screen this event from further analyses.	NA	
A.4.b - Raise the transfer cask with the loaded MPC until it just breaks the SFP water surface. Rinse the transfer cask exterior surface and disconnect the annulus overpressure system as the transfer cask is slowly raised out of the pool	The relative risk for this activity is almost the same as for positioning the unloaded MPC into the SFP with only two additional considerations that could make the risk slightly higher: the weight of the cask will be higher given that the cask has been loaded with SNF and water and the time the transfer cask is held by the crane is longer due to the washdown process for the transfer cask exterior. For purposes of comparisons to other potential risks, given a cask drop into the SFP, this assessment assumes the SFP develops a substantial leak resulting in a loss of coolant with conditional probability of 0.1 (USNRC, 2014).	1.60E-09	All fuel assemblies present in SFP uncovery per year
A.4.a - Lift the transfer cask enough to clear the SFP wall and lower to approximately 12 inches above the floor	This activity has a higher potential for the cask to impact the wall of the SFP during the lifting over the edge of the SFP compared to the MPC draining activity discussed above. This scenario was considered in Section 5.4.3.2 of NUREG/CR-7016 (USNRC, 2012). For this scenario, an edge drop of the cask on the SFP floor may be more likely than an end drop. An edge drop may have more capability to puncture the SFP although this analysis was not available. The amount of time during which the loaded transfer cask is lifted over the SFP edge is small compared to the much longer time the cask is suspended over the SFP during lowering into and out of the SFP. Further, this handling activity is part of the same heavy lift described above for raising the transfer cask out of the pool. Therefore, this activity is encompassed by the drop frequency assigned above. The SFP capacity to maintain its integrity from this lower drop height is judged to be substantial.	NA	
A.4.c - Move back to the Unit 2 cask washdown area	During this activity, the transfer cask + MPC + SNF + water is moved at the height of about 1 foot above the floor to the CWA. The MPC lid is not yet welded onto the MPC. In the event of a cask tip over, slightly contaminated SFP water could be released from the overturned MPC. If some of the fuel assemblies ruptured during the tip over, then radionuclide release into the FHB would be significant. Reducing the lifting height to one foot will minimize the likelihood of a tip over event unless coupled with a seismic event. Even if a tip over occurred, the release would be inside the FHB. For normal FHB operations, released radionuclides would have to pass through HEPA filters, charcoal filters and the iodine capture system before escaping into the environment. Further, this handling activity is part of the same heavy lift described for raising the MPC from the pool. Therefore, this activity is encompassed by the drop frequency assigned to raising the MPC from the pool.	NA	

Initiating Event Category	Analysis of relative risk	Frequency Events/ Year	Potential for Release
Weld on this first (inner) lid and perform the required weld inspections	During this activity, the transfer cask is held in place by a seismic strap to avoid tip over. A major seismic event would be required to break the strap. Improper lid welding could lead to an increase in the possibility of confinement and/or an MPC structural failure in subsequent moves out to the ISFSI pad and during long-term storage. Loss of confinement could lead to as little as zero release if all the SNF in the MPC is intact. If one or two of the SNF rods has failed during drying and storage, then some of the fission gas from the rods could be released into the MPC cavity. The amount of fission gas would be small enough to not be detectable by area monitors.	NA	
A.4.c - Move the transfer cask to the LPT, lower onto the LPT, and bolt onto the LPT	This operation is similar to previous activities with two major risk- reducing exceptions: (1) for this activity, the SFP water has been removed from the MPC; and (2) the MPC lids are now welded on such that a tip over would most likely not lead to a structural failure of the MPC. The crane does not move the transfer cask above the SFP during this lift.	NA	
	Total fuel uncovery frequency per year for within FHB activities =	3.20E-09 per year	

Table A-2. Potential Initiating Events for Fuel Handling and Storage Outside the Fuel Handling Building

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
A. Heavy load and fuel assembly drops without external influences			
A.4&A.5. Drops or tip overs outside the FHB at the East yard without external influences	Sections 3.3.3 of (PG&E, 2018a) states: As described in Section 4.4, the vertical cask transporter (VCT) must lift and transport either the loaded transfer cask or the loaded storage overpack, including the weight of all necessary ancillary lift devices such as rigs and slings. The loaded storage overpack, being the heavier of the two casks to be lifted, provides the limiting weight for the design of the transporter. Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5%. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the cask transfer facility (CTF), while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack (onto the storage pad)		
A.4.c - Drop from VCT of transfer cask at LPT in East yard	Section 8.2.1.2.1 of (PG&E, 2018a) states: Activity (1) evaluation for: lifting or lowering the loaded transfer cask between its bolted configuration on the LPT and its transport configuration on the VCT. This evaluation shows that based on the minimal height of the lifts and the duration of these activities, the probability of a design basis event during those lifts is not credible. The VCT is a single-failure-proof lifting system for both the cask and canister lifts. A single failure proof crane has a drop probability of 9.6E-8/movement (USNRC, 2001, App 2C). For 80 lifts total for both units this translates to 7.7E-6 drop probability in approximately 10 years, or 7.7E-7/ year. However, the fully loaded transfer cask is elevated only several inches for transport in the VCT. Damage to the MPC or the fuel rods within the transfer cask is not therefore expected, even if the transfer cask used at DCPP was dropped.	NA	
A.5.c - Tip over of VCT loaded with transfer cask at East yard	Section 8.2.1 of (PG&E, 2018a) states: Configuration (1): transfer cask suspended vertically from the VCT on the transport route (a distance of 1.2 miles, 1/3 bedrock, 2/3 surficial deposits over bedrock) between the FHB/AB and the CTF. Road has 600 feet at 8% grade, then 6-8% for another 3000 feet. Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5%. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack on the storage pad The vertical cask transporter lifts the transfer cask used at DCPP to the transport position, a few inches above the LPT. These few inches make it very unlikely the transfer cask is damaged by a tip over and so this event is not included in the assessment.	NA	
A.4&A.5&A.6 - Accidents during transit to CTF without external influences	Section 4.2.3.3.2.4 of (PG&E, 2018a) states: Handling loads for normal and off-normal conditions are addressed in the storage overpack 100 System FSAR, Sections 2.2.1.2, 2.2.3.1, and 3.1.2.1.1.2. The normal handling loads that were applied included vertical lifting and transfer of the storage overpack with a loaded MPC through all movements. The MPC and storage overpack were designed to withstand loads resulting from off-normal handling assumed to be the result of a vertical drop.		
A.4.c - Drop of transfer cask from VCT during transit	Section 8.2.1 of (PG&E, 2018a) states: Configuration (1): transfer cask suspended vertically from the cask transporter on the transport route (a distance of 1.2 miles, 1/3 bedrock, 2/3 surficial deposits over bedrock) between the FHB/AB and the cask transfer facility. Road has 600 feet at 8% grade, then 6-8% for another 3000 feet. The VCT lifts the transfer cask used at DCPP to the transport position, a few inches above the LPT. These few inches make it very unlikely the transfer cask is damaged by a drop and so this event is not included in the assessment.	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
A.5.a - Tip over of VCT with loaded transfer cask during transit	Section 8.2.1 of (PG&E, 2018a) states: Configuration (1): transfer cask suspended vertically from the VCT on the transport route (a distance of 1.2 miles, 1/3 bedrock, 2/3 surficial deposits over bedrock) between the FHB/AB and the CTF. Road has 600 feet at 8% grade, then 6-8% for another 3000 feet. Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5%. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack on the storage pad. The vertical cask transporter lifts the transfer cask used at DCPP to the transport position, a few inches above the LPT. These few inches make it very unlikely the transfer cask is damaged by a tip over and so this event is not included in the assessment.	NA	
3. Drops or tip overs while at CTF without external influences			
A.7.a - Drop of transfer cask onto storage overpack at CTF	Section 8.2.1 of (PG&E, 2018a) states: Configuration (2): transfer cask suspended vertically from the VCT at the CTF (5 feet), prior to being placed atop the storage overpack; i.e., it's bolted configuration on the mating device at the CTF; Section 8.2.1.2.1 of (PG&E, 2018a) states: The transfer cask is aligned directly above the mating device, then lowered onto the mating device and secured. A single failure proof crane has a drop probability of 9.6E- 8/movement (USNRC, 2001, App 2C). For 80 lifts total for both units this translates to 7.7E-6 drop probability in approximately 10 years, or 7.7E-7/ year. The VCT system incorporates a single failure proof crane. The 5-foot drop is minimal, unlikely to result in damage to the transfer cask, MPC, or fuel rods. This event is not considered further in this assessment.	NA	
A.7.b - Drop of MPC onto storage overpack at CTF	Section 8.2.1 of (PG&E, 2018a) states: Configuration (3): transfer cask mounted atop the storage overpack at the CTF and the transporter restrained to the ground. The storage overpack is sitting on the CTF baseplate. Section 8.2.1.2.1 of (PG&E, 2018a) states; After the transfer cask is secured onto the mating device, the MPC is transferred by the VCT into the storage overpack located in the CTF. Section 3.3.3.2.6 of (PG&E, 2018a) provides the Drop Protection Design Criteria; In accordance with NUREG- 0612, prevention of a cask or MPC drop is provided by enhancing the reliability of the load supporting systems by design, using a combination of component redundancy and higher factors of safety than would normally be used for a commercial lift device. Load supporting components include the special lifting devices used to transfer the force of the payload to the cask transporter lift points (including attachment pins, as appropriate), the cask transporter lift points, the overhead beam, the lifting towers, and the vehicle frame. A single failure proof crane has a drop probability of 9.6E- 8/movement (USNRC, 2001, App 2C). For 80 lifts total for both units this translates to 7.7E-6 drop probability in approximately 10 years, or 7.7E-7/ year. The VCT system incorporates a single failure proof crane. Drop of an MPC from the transfer cask into storage overpack during transfer operations of 19 feet would have an MPC damage probability of 0.282 (NUREG-1864, USNRC, 2007). The fuel rod cladding within an MPC has been analyzed to not be damaged for a fall of 20 feet but likely to be failed for a fall of 40 feet. A fuel mechanical damage frequency of 2.2E-7/year is therefore assigned. The damaged fuel within the damaged MPC at the bottom of the CTF would not overheat (HOLTEC, 2012). While the referenced analyses were performed assuming the MPC and the storage overpack were properly mated, the analyses are believed to also be applicable to the dropped MPC case, because the four vents are through the storage overpack which is properly aligned within the CTF.	2.2E-7	1 MPC of assemblies mechanically damaged

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
A.7.b - Tip over of VCT with storage overpack at CTF	Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5%. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack on the storage pad The vertical cask transporter lifts the transfer cask used at DCPP to the transport position, a few inches above the LPT. Since it is qualified not to tip over for strong seismic events, these few inches make it very unlikely the transfer cask is damaged by a tip over and so this event is not included in the assessment.	NA	
A.7.b Drop of storage overpack from VCT into CTF	Section 8.2.1.2.1 of (PG&E, 2018a) states: activity (3) lifting or lowering the storage overpack between the transport configuration on the transporter and entry into the CTF shell. This activity does not involve the movement of fuel, but rather is just in preparation of inserting the MPC into the storage overpack used at DCPP. Therefore, this event is not assessed further.	NA	
4. Drops or tip overs of transfer cask during transit to ISFSI pad without external influences			
A.7.b - Tip over of VCT with fuel loaded storage overpack during transit to ISFSI pad	Section 8.2.1 of (PG&E, 2018a) states: Configuration 4: storage overpack being transported to the ISFSI storage pad, suspended vertically from the cask transporter. In terms of seismic stability, this configuration bounds configuration (2) because the storage overpack is heavier than the transfer cask. Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5 percent. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack on the storage pad. Procedures limit the height of the loaded storage overpack to 10 inches above the ground. This height of 10 inches makes it very unlikely that the VCT loaded with a storage overpack would top over during transit to the ISFSI pad (PG&E, 2019g). So this event is not included in the assessment.	NA	
A.7.b - Drop of storage overpack while in transit or onto ISFSI Pad	Section 8.2.1 of (PG&E, 2018a) states: Configuration (4): storage overpack being transported to the ISFSI storage pad, suspended vertically from the cask transporter until anchored on the ISFSI pad. In terms of seismic stability, this configuration bounds configuration (2) because the storage overpack is heavier than the transfer cask. (PG&E, 2018a), The same argument for minimal height of the postulated drop is made here as for transfer cask transport described above. This event is not considered further in this assessment.	NA	
A.7.b - Tip over of storage overpack at ISFSI when anchored to concrete pad	Section 8.2.1.2.3 of (PG&E, 2018a) states: Configuration (5): storage overpack anchored to the ISFSI storage pad in its long-term storage configuration. The storage overpack is designed for high- seismic applications at the Diablo Canyon ISFSI. There is 6 feet between storage overpacks, and the CTF is 100 feet from the concrete pads; storage overpack center to center distance once installed at the pads is 17 feet. Figure 4.2-2 of (PG&E, 2018a) cask concrete embedded rods are 83 inches into concrete. The results in Section 8.2.1.2.3.1 for seismic events clearly show that even for a strong seismic event the storage overpack would not tip over while anchored to the ISFSI pad. Without external influences that are considered later, there is no reason to consider this event further.	NA	
5. Tip over or collision of the VCT loaded with a transfer cask/MPC			
A.5.a. - While cask moved	Section 3.3.3.2.9 of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a), states that the vertical cask transporter is equipped with a cask restraint to secure the cask during movement.	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
A.5.b. - When impacted by vehicle	The VCT has a limited exposure time on the road to other vehicles when carrying fuel; i.e., typically just 10 to 12 hours per year for a campaign of four MPCs. Per Section 8.2.6 of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a), page 8.2- 41, vehicle movement is controlled in the vicinity of the VCT when it is transporting fuel. Therefore, this category of events is not assessed further in this study.	NA	
A.5.a. - When error while driving or driver incapacitated	The VCT can operate in drive or hoist mode but not both, as governed by a selector switch. When the VCT is transferring fuel to the CTF, a support team of two persons is walking along both sides of it at the same slow pace of up to 0.4 mph. (PG&E, 2018a, Table 3.4-4) In the event of a driver error or incapacitation that directs the VCT off-course, either of these support team members can observe that the driver is in distress or acting erratically and make use of one of the two external stop switches, mounted one on each side of the VCT, to bring it to a halt. Therefore, this category of events is screened from assessment in this study.	NA	
A.5.a. - Transporter engine or brake failure	VCT engine failure while transferring fuel would be detected by the halt of any engine-driven activity taking place at the time. A hydraulic fluid leak would be detected by the pressure instrumentation in the hydraulic system and possibly by visual observation of leaking fluid. In the event of a VCT engine failure, the system automatically applies brakes or stoppage of the load movement by engagement of mechanical locks. Per Section 4.3.2.1.2 of (PG&E, 2018a): The cask transporter is also equipped with an automatic drive brake system that applies the brakes if there is a loss of hydraulic pressure (e.g., spring set tractor motor brake) or decelerates if the drive controls are released (e.g., hydraulic system pressure relief). Additionally, the fully loaded cask transporter is not capable of coasting on a 10 percent downward grade even with the brakes disengaged, due to the passive resistance in the drive system that is inherent in the design of each multi-stage, planetary gear, tractor drive transmission. Section 4.3.2.1.2 of (PG&E, 2018a) states: Therefore, the transporter design includes fail-safe features to automatically stop the vehicle (when moving) or stop load movement and apply mechanical locks to hold the load (when lifting or lowering) if the operator is incapacitated for any reason. Therefore, this event is not assessed further.	NA	
A.6. - Transporter fire	Per Section 8.2.5.2 of the DCPP ISFSI UFSAR (PG&E, 2018a) which states: "The fuel tank capacity of the onsite transporter is limited by the Diablo Canyon ISFSI Technical Specifications (TS) to a maximum of 50 gallons of (diesel) fuel. This same section also notes: “Section 11.2.4 of the storage overpack 100 System FSAR presents an evaluation of the effects of an engulfing 50-gallon fuel fire for both overpack and transfer cask. Results of these analyses indicate that neither the storage cask nor the transfer cask undergoes any structural degradation and that only a small amount of neutron shielding material (concrete, Holtite-A, and water) is damaged or lost. This analysis bounds any onsite, 20-gallon vehicle-fuel-tank fire (Event 2)." Therefore, this category of events is screened from assessment in this study.	NA	
A.6. Loss of transfer cask supplemental cooling	Section 8.2.17 of (PG&E, 2018a); SUPPLEMENTAL COOLING SYSTEM (SCS) FAILURE states: SCS system is a supplied fluid device used at DCPP to provide supplemental transfer cask cooling during the loading operation of high burnup fuel while utilizing temporary shielding on the transfer cask, and unloading operation of any MPC. Because the SCS is a keep full system, the only failure mode is a complete loss of annulus water from an uncontrolled leak or line break. Section 8.2.17.2 of (PG&E, 2018a) states: “For the condition of a vertically oriented transfer cask with air in the annulus, the maximum steady state temperatures are below the accident temperature limits for fuel cladding and components (Reference 63).” Therefore, this event is not considered further.	NA	
C.5.b. - Loss of offsite power	Section 4.3.3 of (PG&E, 2018a), page 4.3-8 states: As the transporter ascends the hill along Reservoir Road, it passes beneath the Unit 2 500-kV transmission lines, which are approximately 55 feet above the road surface. To ensure there remains an electrically safe working distance between the transporter and the transmission lines, the maximum height of the lifting beam on the transporter will be administratively controlled in accordance with plant procedures. It is very unlikely that there would be a dropped AC power line in the absence of external influences. This event is therefore not assessed further. Offsite power is the only supporting system used for the transfer cask and systems. Per Section 4.4.4.1 of (PG&E, 2018a) of the Diablo Canyon ISFSI UFSAR: electric power is required to support functions of the mating device hydraulic and airbag systems. Electric and pneumatic power supplies are provided for the mating device hydraulic and airbag systems. Normal power is supplied from the non-safety-related 12-kV distribution system and 12kv/480v transformers for the CTF and for the storage-pad-area normal lighting. All lifting of the storage overpacks and transfer casks are performed by the vertical cask transporter which is powered by an onboard diesel engine.	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
	The transporter also performs the downloading of the MPC into the storage overpack. From Section 8.1.6.3.2 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a); at the CTF the mating device hydraulic and airbag systems require electric and pneumatic power to operate. If the mating device hydraulic system is in Operation at the time of loss of electrical power, the hydraulic system can be operated manually. If the mating device airbag system is in operation at the time of a loss of pneumatic power, removal of the MPC from the transfer cask will be restricted until the system is returned to service. Maintaining the MPC in the transfer cask is considered the safest condition. There are no functions at the CTF related to safe operation of the ISFSI that are electrically powered. Therefore, this category of events is not considered further in this study.		
E. External events (resulting in impacts, losses of inventory, or losses of SFP cooling)	Table 3.4-1 of (PG&E, 2018a) presents external hazard design criteria.		
E.1. - Seismic; General information on handling and storage failures evaluated	Section 8.2.1 of Diablo Canyon ISFSI UFSAR (PG&E, 2018a) states: Cask handling activities outside the FHB/AB were reviewed to identify potential risk significant configurations during a seismic event. Section 8.2.1.2 of (PG&E, 2018a, page 8.2-3) states: PG&E developed the ISFSI Long Period (ILP) earthquake spectra to be used for the analyses of transporter stability, slope stability, and ISFSI storage pad sliding to provide extra design margin since these analyses' results could be affected by long period energy. Page 8.2-5 of (PG&E, 2018a) states: PG&E evaluated the risk of an earthquake causing ground accelerations twice those of the ILP occurring simultaneously with VCT activities (in progress approximately 12 hours per year) and concluded that the risk is not credible (less than 1E-7 per year).		
E.1 – Seismic drop of transfer cask with the loaded MPC while it is above the SFP for decontamination	The FHB steel super-structure which supports the bridge crane has been assessed for seismic events only for conditions when the crane is in use; the parked position being much more seismically robust. It was found that the seismic initiated failure of the FHB super-structure has a failure frequency of 2.89E-5 per year. The FHB crane would not be in use for much of the time. The product of the seismic failure frequency, 2.89E-5 per year, times the fraction of time the FHB crane is in use and above the SFP (~5 minutes for each fuel assembly movement) is approximately 0.006 fraction of a year, and a conditional probability of SFP leakage given the crane drops which should be much less than 0.1, used for a fully loaded transfer cask used at DCPP drop, is judged to be very small (i.e., <1E-8/year) for seismic events, and so neglected in this assessment. The crane is in use to lift and move a loaded transfer cask while it is being decontaminated over the SFP. This is about 3 hours for each MPC transferred to the ISFSI. Uncovery of fuel assemblies in the SFP caused by seismic failure of FHB crane and drop of transfer cask being moved at time of event (total for both Units 1 and 2; 80 MPCs moved averaged over a 10 year) is 8.54E-9/year. This frequency is considered in the analysis. This analysis of the frequency is considered bounding. In PG&E's analysis of the seismic capacity (PG&E,2019h), it was judged that for the crane to drop it would likely require a roof truss failure that results in the columns spreading out (several feet) to allow the crane to slide off its support structure. It is not clear, therefore, that this failure mechanism would realistically result in a drop of the bridge (with a loaded crane) straight down onto the pool deck or angled to drop into the pool.	8.54E-9	
E.1 - Seismic: Drop from VCT of transfer cask at LPT	Section 8.2.1.2.1 of (PG&E, 2018a) states: activity (1) evaluation for: lifting or lowering the loaded transfer cask between its bolted configuration on the LPT and its transport configuration on the transporter. This evaluation shows that based on the minimal height of the lifts and the duration of these activities, the probability of a design basis event during those lifts is not credible. For an exposure of 10 to 12 hours per year, the frequency of a double design earthquake (DDE) seismic event (i.e., at 1.75g SA) during this time would be only 1E-8/year and much less than that for >6.0g SA. See Table 5.1.1-1 of (PG&E, 2018b). Therefore, this event was not considered further.	NA	
E.1. - Seismic: Tip over of VCT loaded with transfer cask at East yard	See assessment for: "Tip over of VCT with loaded transfer cask during transit" below. This event is also not considered further in this assessment. Section 8.2.1.2.1 of (PG&E, 2018a, page 8.2-5) states: The analysis determined that the VCT would not overturn or leave the roadway from a seismic event (Configurations 1, 2, and 4, and a portion of Configuration 1).	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
E.1. - Seismic: Drop of transfer cask from VCT during transit	Section 8.2.1 of (PG&E, 2018a) states: Configuration (1): transfer cask is suspended vertically from the vertical cask transporter on the transport route (a distance of 1.2 miles, 1/3 bedrock, 2/3 surficial deposits over bedrock) between the FHB/AB and the cask transfer facility. Road has 600 feet is at 8% grade, then 6-8% for another 3000 feet. The VCT lifts the transfer cask to the transport position, a few inches above the LPT. These few inches make it very unlikely the transfer cask is damaged by a drop and so this event is not included in the assessment.	NA	
E.1. - Seismic: Tip over of VCT with loaded transfer cask during transit	Section 8.2.1.2.1 of (PG&E, 2018a) states: For all ILP design earthquakes, the analysis showed the VCT would not overturn or leave the roadway; if VCT is carrying the transfer cask horizontally (this position is claimed to be bounding to vertical), then even at 2 times the acceleration would not overturn or leave roadway whose width is >22 feet, versus 18 feet VCT width. VCT transport is an average of about 10 to 12 hours per year. Page 8.2-8 states: the maximum VCT horizontal excursion (transverse to road) was evaluated to be 10.7 inches (roadway minimum width is 22 feet and the width of the transporter from outside of track to outside of track is approximately 18 feet. So if VCT is located directly in middle of the road, it would have about 14 inches on either side after sliding transversely). The maximum VCT slide if aligned along the road at 8.5% grade is 30.2 inches, much less than length of the VCT track. The VCT is 27.5 feet in height. Two times the DDE design earthquake acceleration (3.5g SA) would be 7.0g SA with a seismic frequency of <6E-7/ year. The hours spent in this activity are only approximately 10-12 hours per year so the combined frequency of a larger seismic event and the transit activity taking place is less than 1E-8/year. There is also some probability no damage would occur to the transfer cask even given it tips over. This event is not considered further in this assessment.	NA	
E.1. - Seismic: Drop of transfer cask onto storage overpack at CTF	Section 8.2.1 of (PG&E, 2018a) states: Configuration (2): transfer cask suspended vertically from the VCT at the CTF (5 feet), prior to being placed atop the storage overpack; i.e., its bolted configuration on the mating device at the CTF; Section 8.2.1.2.1 of (PG&E, 2018a) states: The transfer cask is aligned directly above the mating device, then lowered onto the mating device and secured. The 5- foot drop is minimal and the time spent in this operation is limited. The time spent would have to exceed 90 hours per year to achieve a 1E-8/ year frequency of a DDE while in this configuration and then there is still the less than 1.0 probability that the transfer cask is damaged by the drop. This event is not considered further in this assessment.	NA	
E.1. - Seismic: Tip over of VCT with transfer cask at CTF	Section 8.2.1 of (PG&E, 2018a) states: Configuration (1): transfer cask suspended vertically from the VCT on the transport route (a distance of 1.2 miles, 1/3 bedrock, 2/3 surficial deposits over bedrock) between the FHB/AB and the CTF. Road has 600 feet at 8% grade, then 6-8% for another 3000 feet. Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5 percent. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded storage overpack from the CTF to the storage pad, or while moving a loaded storage overpack on the storage pad. This event is not included in the assessment.	NA	
E.1. - Seismic: Drop of storage overpack from VCT into CTF	Section 8.2.1.2.1 of (PG&E, 2018a) states: activity (3) lifting or lowering the storage overpack between the transport configuration on the transporter and entry into the CTF shell. The time spent in this activity would have to exceed 90 hours per year to achieve a 1E-8/ year frequency of a DDE seismic event while in this configuration and then there is still the less than 1.0 probability that the storage overpack is damaged by the drop. This event is not considered further in this assessment.	NA	
E.1. - Seismic: Tip over of VCT with storage overpack during transit to ISFSI pad	Section 8.2.1 of (PG&E, 2018a) states: Configuration (4) storage overpack used at DCPP being transported to the ISFSI storage pad, suspended vertically from the cask transporter. In terms of seismic stability, this configuration bounds configuration (2) which is applicable here because the storage overpack used at DCPP is heavier than the transfer cask. Later in Section 8.2.1 it states, “After the MPC transfer operation is executed at the CTF, the cask transporter lifts the loaded overpack out of the CTF and carries the loaded overpack in a vertical orientation to its final position on the ISFSI storage pad. During this transport a seismic strap is secured to the overpack.” Section 3.3.3.2.5 of (PG&E, 2018a) states: The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5 percent. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded overpack from the CTF to the storage pad, or while moving a loaded overpack on the storage pad. This event is not included in the assessment.	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
E.1.- Seismic: Drop of storage overpack while in transit or onto ISFSI Pad	Section 8.2.1 of (PG&E, 2018a) states: configuration (4): storage overpack as used at DCPP being transported to the ISFSI storage pad, suspended vertically from the cask transporter. In terms of seismic stability, this configuration bounds configuration (2) because the storage overpack 100SA overpack is heavier than the transfer cask. Section 8.2.1.2.1 of (PG&E, 2018a) states: the vertical cask transporter lifts the loaded overpack (storage overpack) out of the CTF and carries the loaded overpack in a vertical orientation to its final position on the ISFSI storage pad. During this transport a seismic strap is secured to the storage overpack. The time spent would have to exceed 90 hours to achieve a frequency of a seismic double design earthquake while in this configuration and then there is still the less than 1 probability that the storage overpack is damaged by the drop. This event is not considered further in this assessment.	NA	
E.1. - Seismic: Tip-over of storage overpack at ISFSI when anchored to concrete pad	Section 8.2.1.2.3 of (PG&E, 2018a) states: Configuration (5): storage overpack used at DCPP anchored to the ISFSI storage pad in its long-term storage configuration. Section 8.2.1.2.3, Seismic Analyses of the storage overpack used at DCPP anchored to the ISFSI Storage Pad in its Long-Term Storage Configuration, states: The storage overpack design basis deceleration level is 45g when tipped over; so even if it tipped over, it would maintain its geometry. Page 8.2-17: the safety factor against sliding is 1.39; so the cask design acceleration is ok. Page 8.2-19: up to three anchor studs could lose pretension and the cask, anchorage, and storage pad would (results in Section 8.2.1.2.3.2) remain within their original design basis load limits. Concrete pad sliding - assessed for design earthquakes to slide at most 1.21 inches. These design calculations were performed for seismic accelerations defined by the Hosgri fault; i.e., 2.1g SA. The design calculations indicate there is much margin above the 2.1g SA evaluated. For this assessment, in the absence of calculations for seismic events above 2.1g SSA, it is judgmentally assumed that tip over or damage to the anchored storage overpack would not occur up to 3.5g SA; i.e., which defines a HCLPF for the storage overpack. The frequency of seismic events greater than 3.5g SAS at DCPP has an exceedance frequency of 9E-6 per year. Earthquakes greater than 6.0g SA have exceedance frequencies of 6.4E-7 per year (PG&E, 2018b). The frequency of damage to an anchored storage overpack due to seismic events, assuming a HCLPF of 3.5g SA defines its mean fragility curve (e.g. the probability of failure above 6.0g SA is then 0.25), yields an integrated failure frequency over all accelerations of 2.8E-7 per year. It seems prudent to expect potential damage to the storage overpack anchored configurations at these much stronger earthquakes. The number of storage overpacks damaged in a single, strong seismic event, among the 137 present when dry storage is at full capacity, depends on the degree of correlation for these seismic failure modes between the 137 storage overpacks installed at the ISFSI. This assessment assumes five storage overpacks would be tipped over or severely damaged which corresponds to fewer fuel assemblies than the equivalent to one reactor core. For seismic events just above 3.5g SA, it is most likely only one storage overpack, would be affected; whereas for much larger seismic events, many may be affected. Since a detailed analysis of this seismic failure mode has not been performed, it is not possible to say whether the "damage" envisioned would be simply mechanical damage of the fuel rods leading only to radioactivity released from the gaps between the fuel and the cladding, or would involve substantial degradation of the passive ventilation system. The four vents in the storage overpack design make it difficult to see how they could all be choked off in such a way to eventually lead to fuel overheating. The release from fuel overheating, if feasible, would still be limited by the extended decay times for fuel assemblies in the ISFSI, many of which were removed from the reactors more than 10 years previously.	2.800E-07	Average of 5 storage overpacks tipped over obtained by weighting frequency of failure by increasing # tipped at higher acceleration ranges; bounded by saying fuel overheats for all that tip over
E.2. - Aircraft crashes	Per Section 2.2.1.3 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a, page 2.2-9), the current frequency of an aircraft crash into the ISFSI is just 8.26E-7/year, with nearly all of the frequency attributable to smaller, general aviation aircraft. Given an aircraft crash into the ISFSI, a conditional damage probability of one or more storage overpacks is assumed to be 1.0. Section X.1.6.3 of the reactor at-power calculation for external events (PG&E, 2016a) lists several conditional reactor core damage probabilities for class 1 structures that vary from 0 to 1.0. Therefore, the 8.26E-7/year frequency is judged qualitatively appropriate for the storage overpacks. The fuel in affected storage overpacks is conservatively assumed to overheat if impacted.	8.26E-07	5 MPCs effected and that fuel overheats
E.3. - Meteorite impact	An asteroid of about 20 meters in diameter, like the one that struck Chelyabinsk, Russia, in 2013 (20 meters in diameter) which may strike the earth twice a century (Planetary Society, 2018), would have a probability per year of striking a 10 square mile area of just 3E-10/year. This is a very small likelihood to hit either of the DCPP and ISFSI sites and so is neglected for this assessment.	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
E.4 - Explosions	Per Section 8.2.5.2 of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a, page 8.2-36), all stationary fuel tanks are at least 50 feet from the ISFSI storage pad security fence and at least 100 feet from the transport route and the CTF. Section 8.2.6 of (PG&E, 2018a) of the UFSAR for the DCPP ISFSI considered seven different sources of explosions including sources involving an onsite vehicle fuel tank and the bulk hydrogen storage facility setback from along the route from the FHB to the ISFSI. Administrative controls are imposed to ensure that no fuel sources exceed the bounding conditions of the analyses that have been performed. Section 8.2.6 concludes that the effects of Diablo Canyon site explosion events involving detonation are enveloped by the FSAR analysis of design-basis accident conditions (explosion and transfer cask side drop) for the storage overpack design used at DCPP, or are not considered risk significant in accordance with Regulatory Guide 1.91. Further, Section 2.2.1.2 notes that there are no commercial or explosive materials stored within 5 miles of the site. Therefore, this category of events is screened from assessment in this study.	NA	
E.5. - Pipeline explosions	Per Section 2.2.1.2 of the UFSAR of the ISFSI (PG&E, 2018a), there are no natural gas or other pipelines that pass within 5 miles of the site. Therefore, this category of events is screened from assessment in this study.	NA	
E.6. - Truck or railcar explosions	Per Section 2.2.1.2 of the UFSAR of the ISFSI of (PG&E, 2018a), there are no rail lines or transportation routes within 5 miles of the ISFSI. Therefore, this category of events is screened from assessment in this study.	NA	
E.7 - External flooding	Per Section 2.4.2 of the UFSAR for the ISFSI of (PG&E, 2018a), there are no dams in a position that could flood the ISFSI site which is at 310 feet in elevation. The roadway provides an overflow path for intense precipitation. Failure of the adjacent reservoirs would drain the water away from the concrete pads. The transporter route from the FHB to the ISFSI is at a minimum of 80 feet above mean sea level. Therefore, this category of events is screened from assessment in this study.	NA	
E.8. - External fires	Per Section 8.2.5.2 of the UFSAR of the ISFSI of (PG&E, 2018a, page 8.2-36), "Administrative controls are imposed to ensure no combustible materials are stored within the security fence around the ISFSI storage pads. Prior to any cask transport, a walkdown will be performed to ensure all local combustible materials (Event 4), including transient combustibles, are controlled in accordance with ISFSI fire protection requirements." "The native vegetation surrounding the ISFSI storage pad is primarily grass, with no significant brush and no trees. Maintenance programs prevent uncontrolled growth of the surrounding vegetation. A mineral oil fire in the Unit 2 transformers, which are located 240 feet from the transporter route at its closest point, has been evaluated and found to be bounded by the analyses performed for a ruptured 2000-gallon gasoline tanker truck, which determined that at a distance of more than 100 feet it does not result in exceeding the design basis of the transfer cask. Further, administrative controls are in place that will not allow any vehicle motion in the vicinity of the Unit 2 transformers during transport operations. Fuel transport operations are also not permitted when severe weather exists or is predicted. See Section 8.2.5, Fires and Section 2.2.2.2, hazards from fires of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a). Therefore, this category of events is screened from assessment in this study.	NA	
E.9. - High winds	See discussion below for tornado missile strikes, subsumed in that assessment with a 135-mph wind speed at 1E-07/year.	NA	
E.10. - Tornado missiles	The potential impacts of high winds and tornado generated missiles have been considered and described in Section 8.2.2. of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a). The accident analysis for tornado effects involves evaluation of the MPCs assuming bounding missiles sizes and speeds for the evaluations. For: 1) the loaded transfer cask during transport to the CTF (for intermediate missiles at 157 mph a side impact would penetrate the outer shell but not the 3/4-inch inner steel shell), 2) MPC transfer activities at the CTF (each MPC is only exposed for a short time, about 4 hours, and the transfer is not to occur during severe weather so should have very low probability), 3) transport of a loaded storage overpack used at DCPP, lifted just a few inches, to the ISFSI pad (assessed large horizontal missile) , and 4) long-term storage of the loaded overpack at the ISFSI pad. (The anchored cask system, which provides additional resistance to overturning, is bounded by the free-standing overpack analysis; The free-standing overpack is capable of withstanding the combination of tornado wind (or instantaneous pressure drop) and a large-missile-load impact with a conservative safety factor against overturning of greater than two.)	1.00E-07	1 MPC mechanically damaged

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
	No missile strike will cause instability of the overpack, compromise the integrity of the confinement boundary or jeopardize retrievability of the MPC. Considered concurrent wind and missile strikes; A 500kv insulator ring is classified as an intermediate missile at 157 mph and would penetrate the outer shell but not the inner shell of the overpack. The free-standing overpack is capable of withstanding the combination of tornado wind (or instantaneous pressure drop) and a large-missile-load impact with a conservative safety factor against overturning of greater than two. The cask transporter has redundant drop protection by design (Section 3.3.3). Therefore, a loss of MPC load due to a direct missile strike on the transporter is not credible. Per Section 8.2.2.1 describing the event analyzed: "The cause of this event is the occurrence, at or near the ISFSI site, of meteorological conditions that are favorable to the generation of a tornado. The design-basis tornado wind speed for the ISFSI is based on a conservative estimate appropriate for DCPP (annual probability of 1E-7), which was developed by the NRC (Supplemental Safety Evaluation Report No. 7)". Nevertheless, it seems prudent to use 1E-7/year as a frequency bound for greater wind speeds and missile strikes on one storage overpack cask anchored at the ISFSI pad. This frequency is considered an upper bound on the frequency of unanalyzed events from greater wind speeds. A more realistic assessment would require substantially more effort than is justified by this screening analysis. As a point of comparison, the at-power PRA for DCPP computed a tornado caused reactor core damage frequency of 3.9E-8/year for a sequence involving failure of offsite power (7x10-5/year for 65 mph) and a conditional probability of core damage (i.e., 5.6E-4). Other targets (83 tornado missile targets were considered in the evaluation) were found to yield an additional contribution to the reactor CDF totaling 2E-7/year. The largest individual target contribution was 5.22E-8/year. The impact of this individual target was to take out the 480v switchgear ventilation resulting in a conditional reactor core damage probability = 1.1E-2.		
E.11. - Turbine missiles	Not applicable to ISFSI because it sits on a hill well above the DCPP turbine building. The time spent in transit by the vertical cask transporter from the fuel handling building to the ISFSI is a small portion of the total time while the reactors and their turbines are operating. The combined probability of a turbine missile with this small, time fraction for transit time is very small.	NA	
E.12. - Security/sabotage/terrorism events	This category of initiating events is not considered within the current assessment. Such assessments are typically classified as they require knowledge of existing defenses against such attacks. The assessment of the frequencies of such attacks is also problematical.	NA	
E.13. - Onsite or nearby chemical releases	Per Section 2.2.2.4 of the UFSAR for the ISFSI of (PG&E, 2018a), an administrative program is to control hazardous chemicals in the ISFSI storage and CTF areas and along the transportation route. Further, per Section 8.3, "there are no potential onsite fires, explosions, or chemical hazards that would have a significant or unacceptable impact on the ISFSI." Therefore, this category of events is screened from assessment in this study.	NA	
E. 14. Extreme external temperatures	Per Section 8.2.10.1 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a), an extreme high environmental temperature is 125°F. The maximum temperature experienced at the Diablo Canyon site is 104°F. The analysis for a storage overpack installed at the pad showed that with the environment fixed at 125°F until the components reached equilibrium showed that all components are well below the accident-condition (short term) design-basis component temperatures, and that the MPC internal pressure was found to be no greater than 84.3 psig as compared to the accident design internal pressure of 200 psig. Therefore, this category of events is screened from assessment in this study.	NA	
E.15. – Ocean shipping	Per Section 2.2.1.2 of the DCPP ISFSI UFSAR (PG&E, 2018a): "Local shipping tankers may come within 10 miles of the DCPP site, but will remain outside of a 5-mile range. Coastal shipping lanes are approximately 20 miles offshore. Therefore, shipping does not pose a hazard to the ISFSI." Section X.1.14 of the DCPP PRA for external events (PG&E, 2016a), also considered the potential for ships impacting the intake cove, and from ship impacts blocking the flow of water into the intake cove. A loss of auxiliary saltwater (ASW) was computed to have a frequency of 2.9E- 6 per year from such events. These events can impact SFP cooling but not the ISFSI located well above the DCPP. These frequencies are well below the frequency of other causes of losing SFP cooling and so are not considered further.	NA	
F. Criticality events			
F.3. - Loading of an unauthorized fuel assembly into an MPC	The limiting reactivity condition occurs in the SFP during fuel loading, where assemblies are loaded into the MPC in close proximity to each other, with moderator between assemblies. From Section 8.2.9.2 of the Diablo Canyon ISFSI UFSAR (PG&E, 2018a), "The chance of loading of an unauthorized	NA	

Initiating Event Category	Notes and discussion as to applicability of the potential initiating event	Frequency Events/ Year	Potential for Release
	fuel assembly is greatly minimized because of the multiple administrative controls imposed via procedures to ensure a fuel planning or loading error does not remain undetected". As an example, "...the spent fuel loading procedures require that a final verification of the identity and location of fuel assemblies be performed prior to placing the lid on the MPC." Per Section 4.2.3.3.5 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a, page 4.2-26), neutron absorbing panels are included in the storage overpack fuel basket structure to assist in the control of reactivity. In its storage configuration, the storage overpack used at DCPP is dry (no moderator), and the reactivity is very low (k_{eff} less than 0.515). Therefore, this category of events is screened from assessment in this study.		
G. Loss of heat removal from storage overpack installed at ISFSI Pad			
G.1. - Blockage of ventilation ducts	Metal screens on the inlet and exhaust of each ventilation duct prevent foreign objects, such as insects and avian activities, from falling in. These metal screens are visually inspected every 24 hours; see Section 8.1.4.4 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a). Per Sections 8.1.4 and 8.2.13 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a), even postulating blockage of two of the four ventilation ducts on a given storage overpack installed on the pad, heat removal would still be sufficient to meet component temperature limits and the MPC accident design internal pressure limit of 200 psig. The only identified event which could conceivably block all four ventilation ducts of a storage overpack over cask at DCPP is that of a land slide or tornado, which could pile debris surrounding one or more storage overpacks installed on the pads. These events are discussed further as potential adiabatic heatup initiator below.	NA	
G.2. - Adiabatic heatup	An envisioned cause of blocking all four ventilation ducts in a storage overpack installed at the pad at Diablo Canyon is that of debris from a tornado. The results of an adiabatic heatup analysis are documented in Section 8.2.15.2 which showed that both the fuel-cladding and the MPC-confinement boundary temperatures remain below their respective short-term-temperature limits up to at least 72 hours (i.e., the end of the analysis), or 1058°F/570°C) by over 150°F, and the confinement boundary by almost 175°F. The MPC Internal pressure limit of 200 psig is also calculated to be just 90.5 psig and so the limit is met. It is concluded that at least 72 hours is available to clear the blockage before fuel cladding or confinement boundary limits are exceeded. According to Section 8.2.2.1 of the Diablo Canyon ISFSI UFSAR of (PG&E, 2018a), the design wind speed for DCPP was selected for a conservatively calculated annual exceedance frequency of 1E-7 per year. Per Figure 8-17 of the Diablo Canyon PRA calculation for other external events (i.e., Calculation File No. X.1 Revision 1, DCPP Other External Events), (PG&E, 2016a) this frequency for strike by a tornado at DCPP corresponds to a wind speed of 130 mph or greater. The tornado strike frequency for the low end of tornado wind speeds (i.e., 65 mph) by comparison is shown as 7E-5 per year in the same Figure 8-17. It is judged that it would require a rather strong tornado to create and carry enough debris to block the ventilation ducts on three or more sides of a storage overpack used at DCPP and installed at the ISFSI. A 130-mph wind speed seems reasonable and so a frequency of a tornado initially blocking the ducts is taken to be 1E-7 per year. The ISFSI is far enough away that the chance of fully blocking the vents of one storage overpack at DCPP is judged to have no more than a 10% chance, but conservatively, this potential 10% reduction factor is not credited. There would also be the possibility of unblocking the debris within the 72 hours available before temperature limits are exceeded in the storage overpack as used at DCPP with blocked ducts. No credit is taken for removal of the debris from the ducts due to additional debris that may be blocking the access road to the pads. Therefore, a frequency of 1E-7 per year is used for a sequence where a tornado hits DCPP with sufficient wind speeds to carry debris up to the ISFSI, blocks all four ventilation ducts of a storage overpack as used at DCPP, and the debris is not removed prior to fuel cladding failure.	1.00E-07	1 MPC of fuel assem blies eventually overheat
G.2. - Adiabatic heatup	Another envisioned cause of blocking all four ventilation ducts in a storage overpack installed at the pad at Diablo Canyon, is that of a seismic initiated landslide large enough to bury some of the storage overpacks. See the discussion in Section 4 for potential initiating events in Group G. Not all storage overpacks would be impacted by a landslide at 1E-7 per year and not all those impacted would necessarily overheat sufficiently to release cesium. Such a slide, though quite unlikely, could reasonably impact multiple storage casks. 20 storage overpacks each containing a fully loaded MPC are assumed to be buried sufficiently to block all of their vents.	1E-7	20 MPCs of fuel assemblies eventually overheat

Appendix B

Key Assumptions in this Assessment

Table B-1. Summary of Assessment Assumptions

ID	Risk Contributor	Statement of Assumption	Basis for Assumption	Impact on Results
1	Selection of Initiators	The initiators governing the risk from the SFP at DCPP are **seismic events beyond the design basis** (i.e., greater than 1.75 g spectral acceleration), and **heavy load drops impacting the SFP.**	This assumption is based on conclusions from previous SFP risk studies. For DCPP, the seismic hazard exceedance frequency is higher than for the plants analyzed previously, and the hazard frequency slope does not decrease as rapidly at higher acceleration levels. Further, seismic events are the dominant contributors to large, early releases for DCPP reactor core damage accidents. Heavy load drops impacting the SFP are one way to mechanically impact the SFP. Much time is available for the operators to respond using available procedures, to loss of SFP cooling events from other causes or to smaller losses of coolant from other causes; i.e., other than by seismic events.	Internal plant fires are a major contributor (i.e., about 50%) to DCPP reactor core damage frequency, though much of the fire contribution comes from fires in the turbine building. Fires and internal floods initiating in the fuel handling building and adjacent auxiliary building rooms could impact SFP cooling and makeup systems. These spatial impact-oriented initiators may require further assessment.
2	Direct Seismic Impact on SFP	The **seismic failure of the auxiliary building,** which has a high confidence (95%) of low probability of failure (<.05) (HCLPF) of 2.33 g spectral acceleration (SA) and median capacity of 4.74 g SA, is modeled as a loss of cooling geometry for the SFP with conditional probability of 1.0; i.e., no credit is taken for coolant inventory makeup from within or adjacent to the SFP area. It is possible that spray over the FHB using a portable crane from outside the failed structure would still be feasible, but this is judged not to prevent fuel uncovery or damage. It could successfully limit the release offsite.	The AB seismic capacity was developed for the DCPP Long Term Seismic Program. Collapse of the AB may result in debris falling into the SFP as well as onto SFP cooling equipment within the building. It could also obstruct access to the SFP area. Its collapse would certainly complicate actions to provide makeup to the SFP in time to prevent fuel uncovery, including even to determine the SFP coolant level with time.	The failure of the AB structure contributes to the total seismic frequency of uncovering the fuel in the SFP. The specific auxiliary building seismic failure mode assessed in its evaluation should be reviewed to more closely examine its potential impact on the FHB areas and SFP cooling equipment.
3	Direct Seismic Impact on SFP	Seismic **capacity of the SFP concrete structure** is assigned the same seismic capacity as developed by the DCPP Long Term Seismic Program for the ASW intake structure; i.e., a HCLPF of 3.5 g SA and median capacity 0f 7.65 g SA. Its seismic failure is modeled as a loss of coolant event of two different equivalent hole sizes; small (1.5 inch, 50% of failures) and moderate (6 inch, also 50% failures). Credit is given for operator recovery actions only for the small size breaks. The moderate size break is sufficient in size to relatively quickly drain the SFP. The loss of coolant to other areas in the FHB/AB would further complicate any operator actions within the building. Actions outside the FHB to align for external spray may still be feasible to mitigate radiological releases, but would not prevent fuel uncovery.	The SFP seismic structural capacity was not developed in the DCPP Long Term Seismic Program. PG&E subsequently estimated its seismic HCLPF as being 3.5 g SA. A median capacity of 7.65g SA is assigned for this study. A seismic capacity curve shape was chosen with the same HCLPF to represent the SFP structure. Since a detailed seismic capacity curve beyond design basis level acceleration is not available, a bounding interpretation of failure impact is used.	The failure of this structure, which is seismically robust, contributes marginally to the frequency of uncovering the fuel in the SFP. Its contribution would be higher only if additional credit for operator recovery actions is judged to be realistic.

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4	Direct Seismic Impact on SFP	Human error probabilities (HEPs) for actions outside the control room in response to a seismic event were assigned subjectively; e.g., for restarting the SFP cooling pumps after a loss of offsite power, and for aligning for alternate paths to restore SFP cooling or SFP coolant makeup. The HEPs are consistent with those assumed by EPRI for low seismic conditions (EPRI, 2014). These **human error probabilities** were increased by a factor 10 for higher accelerations between 1.75g SA and 3.0g SA, and then to 1.0 for seismic accelerations greater than 3.0g spectral acceleration.	The actions to align for alternate paths to restore SFP cooling and/ or coolant makeup to the SFP, including FLEX actions, are directed by procedures at DCPP. Setting the HEPs to 1.0 at very high accelerations is consistent with the modeling approach used in the DCPP reactor core damage studies. Above 3.0 g SA, if one of the reactor units is initially operating, there is a 95% chance it is concurrently undergoing core damage and a 40% chance that a large, early release is in progress. AC power at the 480v level very likely is also lost. Control Room indications are also more than likely failed as is the fire water system within the auxiliary building. Power for FLEX SFP level indications, even if they survived the seismic shaking, would be discharged within 21 hours.	The assumption of increasing the HEPs at high seismic accelerations has a significant impact on the seismic SFP fuel uncovery frequency; i.e., a factor of 3. However, this change does not change the relative rankings of the risk scores for the four offload scenarios.
5	Direct Seismic Impact on SFP	A **loss of offsite power i**s expected for every seismic event large enough to impact the SFP; i.e., requiring the operators to respond by loading the SFP cooling pumps onto the emergency diesel generators (EDG) as a minimum action.	There is a greater than a 0.5 probability of failing the offsite power supply to the plant for accelerations as low as 0.35 g to 0.5 g SA.	Neglecting the possibility that no operator actions are required at low seismic acceleration is a negligible conservatism. The SFP risks come from higher acceleration ranges.
6	Direct Seismic Impact on SFP	The ability of the **FHB ventilation system** to filter large radiological releases via its HEPA and charcoal filters, after switching to emergency mode, even if there are no seismic structural failures, AC power is available, and there are no hydrogen burns, is judged insufficient to significantly reduce releases offsite from SFP fuel uncovery events. Therefore, the fuel uncovery frequency of such sequences with FHB ventilation working is also included in the totals for all SFP fuel uncovery sequence frequencies. The FHB ventilation is also not credited for seismic sequences in which there are structural failures or when emergency AC power is failed.	It is expected that HEPA filter break through from any large radiological release would occur soon after the onset of fuel release due to overheating in the accident progression, thereby minimizing retention of the radiological release from the uncovered fuel. It is also possible that the FHB would overpressurize, and thereby leak excessively through unfiltered paths for all portions of the accident progression of SFP fuel uncovery sequences.	The frequency of such sequences where ventilation and filtering were initially assessed as successful is only a small fraction of the fuel uncovery sequence frequencies that are unfiltered.
7	Direct Seismic Impact on SFP	A separate seismic capacity evaluation for the SFP liner has not been developed for the DCPP SFP. **Seismic initiated smaller liner leaks** which could be mitigated by operator actions to reestablish SFP cooling or coolant makeup, are represented by the seismic capacity assigned to the SFP structure times 0.5. The other 50% of SFP structural failures are assumed to result in moderate size leaks for which fuel cooling could not be recovered.	The seismic capacity of the DCPP liner alone has not been assessed. It is subsumed by the SFP structural capacity evaluation. This inclusion in the model of small liner leak rates was made to ensure sequences in which the operator actions for makeup were relevant. The assumed seismic capacity for the SFP structure (HCLPF=3.5 g SA) is well above the design basis seismic acceleration of 2.1 g SA.	Because of credit for actions to align for SFP makeup, both normal and emergency water sources when AC power is available, and for FLEX actions when AC power is not available, the frequency of small leaks from SFP structural failures leading to SFP fuel uncovery is very small.
8	Direct Seismic Impact on SFP	The qualitative assessment of SNF risks, **neglects the possible dependence** between the Unit 1 and Unit 2 SFP's responses; i.e., the risk is evaluated for the response of the Unit 2 SFP and the potential release from uncovered fuel assemblies tracked is limited to those fuel assemblies stored at that time in the SFP for Unit 2.	This approach simplifies this assessment. The SFP structures are of very similar design and so are expected to have similar if not identical seismic capacities. However, the offload scenarios are staggered by one year (i.e., the offload scenarios and the final reactor offload occurs 1 year sooner for Unit 1 as compared to Unit 2).	This assumption does not affect the frequency of such seismic SFP failures because the same seismic event is involved. The consequences, would be greater, however, if the fuel assemblies in both SFPs are uncovered in the same seismic event. A conservative assumption might instead be to assume that both SFPs fail seismically together

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				in the same seismic event (i.e., if Unit 2 SFP fails, so does Unit 1 SFP). In that case, the number of fuel assemblies uncovered would be roughly double the amounts assessed here for just the Unit 2 SFP. The USNRC consequence assessments for a generic PWR did consider the joint radiological releases from two shared SFPs and yet still concluded the QHOs would be met with substantial margin.
9	Direct Seismic Impact on SFP	For the fraction of time when the FHB crane is in operation (i.e., to move a fuel assembly or MPC), an additional seismic capacity for the FHB steel super-structure is considered. This failure mode has a HCLPF = 1.485 g SA and median capacity of 2.92 g SA. This seismic capacity only applies when the FHB crane is not parked; i.e., the capacity is weaker when the crane is in operation. The failure of the FHB steel super-structure may allow the crane or its lifted device to fall into the SFP. For this assessment it is assumed that the drop of fuel assemblies would not significantly impact the SFP liner or structure or other fuel assemblies due to their limited weight. A fully loaded transfer cask has a chance of tearing the liner and opening a path for loss of coolant. A 0.1 chance is assumed. See the further discussion in assumption 11 below on such heavy load drops.	The AB and FHB steel super- structure seismic capacities were developed for the DCPP Long Term Seismic Program. No seismic capacity of the DCPP FHB steel super-structure was developed for the case when the crane is parked because it was determined to be much more seismically robust in that condition. Collapse of the AB or FHB steel super-structure may result in debris falling into the SFP as well as onto SFP cooling equipment in the building. The drop of the crane into the SFP, being lighter than a fully loaded transfer cask with MPC (i.e., 117 tons), is not expected to impact the SFP liner and supporting concrete enough to open up a path for a loss of coolant. The amount of time that fuel assemblies are lifted above the SFP racks (but still below the SFP water level), per refueling cycle, is estimated as 1.444 days. Therefore, the fraction of a 1.5-year fuel cycle in which fuel assemblies are being moved is just 0.003.	The seismic failure of the FHB steel super-structure, only applies for the small fraction of the refueling cycle while the crane is in use and lifting a loaded transfer cask with MPC. However, this small combined probability is to be compared with a drop of the MPC from all other causes. This small contribution to the frequency of heavy load drops was included in the assessment and timed to match the time intervals when an MPC is fully loaded for transfer to the ISFSI for dry storage.
10	Effects of Reactor Operation on SFP	A seismic event resulting in a **severe reactor accident and a large, early release** is judged to prevent onsite operator actions needed to mitigate the impacts of the same seismic event on the SFP. The operator actions affected are to align SFP cooling and to align any of a variety of water sources for SFP coolant makeup.	Releases large enough to be considered large offsite and affect public health prior to evacuation are judged likely to also impact onsite operator actions. Nearly all seismic events that could result in severe reactor accidents would also cause failure of offsite AC power stopping the SFP cooling pumps. It's also likely that onsite AC power would also be failed by the seismic event. The PWR pilot SFP study (EPRI, 2014a) identified a related dependency on severe reactor accidents; i.e., such as at Fukushima. There was a concern that hydrogen releases from the reactor containment to the buildings containing the SFP might lead to hydrogen burns or explosions and consequential failure of the SFP cooling systems. However, the pilot study assessment concluded that the impact from hydrogen transport from the reactor containment	This addition to the frequency of fuel uncovery in the SFP is modest; i.e., see Figure 8-8 It has even less effect on the risk comparison between offload scenarios. The EPRI PWR SFP pilot study (EPRI, 2014), considered only the dependence on hydrogen burns from the severe accident in the onsite reactor, and not radiological releases affecting the SFP recovery actions.

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			undergoing a severe accident and combustion within the SFP area was rather small for all containment failure modes. At DCPP, the FHB does not share a common wall with the reactor containment making it even less likely for any hydrogen generated within the reactor containment from entering the FHB. The closest location to the cylindrical containment building is where the fuel transfer tube penetrates both buildings (PG&E, 2017a). Instead, the modeled dependence of operator actions on the large, early releases is thought of as an approximation for all such interactions between severe accidents involving the reactor and the nearby SFP.	
11	Drop of Fully Loaded transfer cask into SFP	**Random Drop** of a TRANSFER CASK with a fully loaded MPC and TRANSFER CASK (~117 tons) causing a loss of coolant in the SFP per movement (i.e., 2E- 9/movement, USNRC, 2001) **into the SFP cask recess area in the SFP**, might be assumed to fail the SFP liner and concrete structure, with probability 1.0. An additional conditional probability of 0.1 is assumed that a loss of coolant occurs given the drop is used here to credit the DCPP specific analyses which concludes there would be no leakage path through the slab even for a bounding 125 ton drop. This approach is consistent with (USNRC,2001). The resulting break is modeled as a loss of coolant event of two different sizes assumed to be of equal probability: a small 1.5-inch equivalent diameter hole and a moderate hole of approximately 6 inches in equivalent diameter. The 1.5-inch hole may be mitigated by recovery actions providing SFP makeup. The 6-inch hole would be large enough to relatively quickly drain the SFP likely precluding operator actions both inside and outside the FHB from mitigating the fuel uncovery; i.e., for the moderate break the failure of makeup recovery actions is assigned a probability of 1.0.	The ACRS review of NUREG-2161 noted that a loaded MPC drop into an SFP was dynamically analyzed on a different nuclear power plant (NPP) as not causing a liner tear. However, the NRC analysts proposed use of a 0.1 probability of tear anyway. A DCPP specific analysis for such an event has been performed as documented in Section 9.1.2.3.11 of the DCPP FSAR (PG&E, 2018), the NRC is quoted as stating: Regulatory Position 5.b- "PG&E has evaluated the drop of a loaded transfer cask from the highest point into the bottom of cask recess area in the SFP at 94 feet 6 inches. The postulated drop is 4.67 feet in air followed by 42.83 feet in water. Analysis demonstrates the adequacy of the affected structures during the postulated drop, demonstrating that the drop will not cause 1) loss of building structural function, 2) damage to the SFP resulting in loss of SFP water, or 3) unacceptable damage to other systems or equipment. The SFP stainless steel liner may be damaged in this drop, however, the structural integrity of concrete forming the SFP is maintained, preventing any uncontrolled leakage". The concrete at the bottom of the SFP is roughly 5 feet thick. The preceding statement was in turn based on the work in (PG&E, 2019f). Section 3.1.9 of that calculation reads as follows: "The consequences of the postulated drop are: a. the rock foundation has the capability of adsorbing energy of impact with an elastic deformation estimated at 1.12 inches. The rock foundation has adequate strength to resist the bearing pressures generated during the impact, b. Concrete slab: The cask is expected to penetrate in the top of the slab about 0.6" along with the liner. No perforation through the slab. However, DCM C-58 formulae indicates some back-face spalling, but this formula is considered conservative for this application. c. Steel liner: Perforation of the liner is indicated, potentially leading to leakage of water and loss of function."	The frequency of a heavy load drop leading to a loss of SFP coolant is so low that eliminating this contribution to SFP uncovery has only a minimal impact on the total risk from the SFP. The added factor of 0.1 credits the DCPP SFP specific analysis indicating an unlikely loss of coolant. This credit is consistent with the USNRC work. The heavy load drop frequency is a generic industry value that has not yet been assessed for the specific design and practices at the DCPP.

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12	Drop Frequency	A **heavy load drop probability** per lift of 9.6E-8 per movement is used for heavy load drops other than over the SFP, because all heavy loads involving SFP or ISFSI handling at DCPP are performed by single failure proof cranes. The frequency of a loss of coolant from the SFP frequency of 2E-9 per movement resulting in a heavy load drop includes additional factors such as the proportion of time over critical areas and SFP damage impact. The frequency of 2E-9/ movement is not applicable to other heavy load drops.	The heavy load drop frequency that causes a loss of coolant from the SFP of 2E-9 per movement comes from NUREG-1738 (NRC, 2001, Appendix 2C), which is based on a fault tree and data analysis. The analysis is for a single failure proof crane, as applies to DCPP, per Section 4.4.1.2.5 of the DCPP ISFSI UFSAR. The assumed drop rate developed by the USNRC is lower than the limited amount of actual experience collected to date for lifts of fuel assemblies (11 out of 344,000 lifts for 3.2E-5/lift) and for heavy load data from 9 plants; i.e., 3 events out of 54,000 lifts for a point estimate frequency of 5.6E-5/lift.	The heavy load drop probability is very uncertain. It would take a factor of 1,000 increase in the heavy load drop per movement probability leading to losses of coolant from the SFP to be of the same order of occurrence frequency as for seismic initiated SFP fuel uncovery sequences.
13	Movement of Fuel Assemblies	This assessment neglects the potential risks from a drop of a fuel assembly on the SFP liner or onto other fuel assemblies. **During a fuel assembly movement** to complete the fuel offload from the reactor, loading of an MPC, or for fuel dispersion (i.e., to rearrange the fuel placements within the SFP), the fuel assembly is raised about 15 feet, though it is still covered by water within the SFP. The time that fuel assemblies are in this elevated position is neglected in so far as it affects the allowable time for operator recovery given a fuel uncovery event form other, concurrent events.	The fuel assemblies are not heavy enough (~1570 pounds) to fail the SFP liner and concrete structure, nor other fuel assemblies should such a fuel assembly drop occur. Several fuel assemblies have been historically dropped without penetrating the SFP liner. The fraction of time that a fuel assembly is elevated 15 feet is limited to just a fraction of the total time the SFP is operating; i.e., 4 to 5 minutes per fuel assembly movement. By a single fuel assembly being elevated, this could reduce the time allowable for the operators to align for SFP coolant makeup due to higher radiation levels in the SFP area and shortening the times until a radiation release occurs; i.e., because there is 15 feet less water covering the one fuel assembly that is lifted. Credit in this assessment for operator actions is only taken if there is not a large hole draining the SFP. This potential impact on operator performance is neglected because the fraction of time a fuel assembly is elevated is not significantly influenced by the different offload scenarios considered.	The number of fuel assembly movements is not significantly affected by the specific offload scenario, so neither is the fraction of time one fuel assembly is elevated. Therefore, the risk differences between fuel offload scenarios should be negligible. The one exception noted is if a seismic event occurs while a crane movement is in progress. The seismic capacity of the FHB steel super structure is reduced while the crane is not parked. This impact is considered as the number of fuel assemblies present at the time of the movement both for refueling and for MPC loading will differ between the different offload strategies.
14	Direct Seismic Impact on Storage Overpack at ISFSI Pad	The **storage overpack mounted at the ISFSI pad is assigned a seismic capacity curve** with a HCLPF of two times the design basis seismic event acceleration, and a seismic capacity curve shape consistent with other robust concrete structures. The failure from such stronger seismic shaking is judged to potentially tip over or crush the storage overpack. The fuel within the MPC may therefore lose natural convection heat removal leading to a possible fuel overheating and radiological release.	The seismic impact on the storage overpack mounted at the ISFSI pads has been evaluated by PG&E for two times the design basis earthquake shaking and found acceptable. The HCLPF acceleration is therefore expected to realistically be higher than this. If the storage overpack is crushed or tips over, the air-cooling by natural convection with vertical flow may no longer work for all four vents. Only two of four vents are sufficient to cool stored fuel, so it is very unlikely tipover could preclude sufficient heat removal.	The contribution from such seismic failures is small compared to other risks evaluated by bounding assumptions involving the dry storage at the ISFSI, and especially small compared to the risks from the SFP.
15	Damaged Fuel, Fuel Debris	The handling of **damaged fuel and fuel debris** does not pose a significant risk in the SFP, handling, or at the ISFSI.	There has not been any damaged fuel loaded transferred to the ISFSI from the SFP. Some fuel failures have occurred and there is a small number of fuel assemblies which have been reconstituted and a few others which currently would constitute damaged fuel if not repaired. These failed fuel assemblies are currently still in the SFP.	The risks posed by damaged fuel and fuel debris should be handled the same and therefore no difference is assessed between the four offload scenarios evaluated.

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			It is expected that what unique risks they pose are the same in all four offload scenarios. The DCPP ISFSI UFSAR (PG&E, 2018a, Section 4.2.3.2.2) says: Any qualified fuel assembly that is classified as damaged fuel may be loaded into an MPC-24E. Up to a total of four damaged fuel containers (DFCs) containing damaged fuel may be stored in a single MPC-24E, with the balance being intact fuel assemblies. Fuel classified as fuel debris must be stored in a DFC and must be loaded into an MPC-24EF. Per Section 10.2.1.1 of the DCPP ISFSI UFSAR (PG&E, 2018a); damaged fuel is inspected when removing the fuel from the reactor and, if not detected before loading an MPC, it can be detected by reactor coolant activity.	
16	Long Term ISFSI Storage	The effects of long-term storage at the ISFSI posed by **corrosion** are judged to be the same for all four offload scenarios evaluated.	The long-term effects would take place over multiple decades and well after the completion of the offload scenarios by 2034 at the latest.	There should be no impacts on the comparative risks between offload scenarios.
17	Radiological Release	A surrogate is used as an indicator for the **amount of radiological release** in any fuel uncovery or fuel overheating sequence postulated. The number of fuel assemblies in proximity to the first fuel assembly damaged or uncovered of its coolant water is used as a representation. For fuel assemblies in the SFP, the total number of fuel assemblies present at the time of the accident and which overheat is used. For drops of transfer casks away from the SFP, the number of fuel assemblies within the loaded MPC that is dropped is used. For loaded storage overpacks mounted at the pads, the number of fuel assemblies within the storage overpack that have lost heat removal is used.	Radiological release can occur from mechanical damage of the fuel cladding, and its protective barriers; e.g., the MPC, its transfer cask or the storage overpack. Accident progressions which lead to overheating of the fuel itself, would involve the greatest releases by fraction of initial inventory, and after the hottest fuel assembly overheats, may propagate to involve all fuel assemblies in proximity. The extent of radiological release can also be a function of the degrees of Zircaloy- steam or Zircaloy air exothermic reactions, which take place. The most important radiological releases after the first few days in the SFP, in terms of potential impacts on human health effects, are expected to be the radionuclides cesium-134 and cesium-137. These radionuclides decay relatively slowly and so the fuel assemblies at risk are approximately proportional to the number of fuel assemblies vulnerable to overheating after adjusting for cesium decay.	For accident progressions in which air-cooling limits the extent of fuel overheating, the amount of radiological release would be less than proportional to the number of fuel assemblies vulnerable to overheating. The radiological inventories available for release diminish as the time after removal from the reactor increases.

Appendix C

Summaries Regarding the Extent of Fuel Damage Given Fuel Uncovery from a Review of the Literature

Some insights obtained from reviews of the available literature organized from oldest to latest dates of publication are now summarized.

1. NUREG/CR-0649 (USNRC, 1979) - In NUREG/CR-0649 (USNRC, 1979) for PWR high density fuel assemblies (i.e., Figure 2d with 10.25 inch fuel cell pitch and with wall- to-wall placement of the fuel racks) with a large baseplate coolant hole (DCPP fuel assemblies have large coolant holes of 5 or 6 inches in diameter) and with a 33 GWD/MTU maximum burnup; shows a minimum decay time to limit peak cladding temperatures to less than 900°C of approximately 2 years. This analysis evaluated the minimum decay times assuming a well ventilated SFP building; i.e., ventilation that is much greater than a normal forced ventilation system could provide. Passive methods involving a "chimney effect" created by opening doors at a low elevation and at a high elevation in the FHB would be needed to achieve the natural convection flows credited across the top of the SFP. DCPP has procedures for opening the needed doors for passive cooling but these steps were not credited in this assessment for strong seismic accelerations. If the fuel assembly racks were not packed in wall to wall, then the decay time may be reduced from about 700 days to about 280 days; i.e., if there is a 12-inch down comer space between the racks and the edges of the SFP (per Table VI). The gaps between the racks and the surrounding SFP walls varies at DCPP, from as small as 4 inches to as large as 13 inches. Also, there is a large cask recess area or "cup area" that is completely open, providing a large vertical column for relatively cool air to reach the bottom of the SFP after coolant is lost before passing up through the heated fuel assemblies. For these reasons the SFPs at DCPP need not be considered a wall-to-wall arrangement. Figures 14 and 16 of NUREG/CR-0649 for a well-ventilated FHB indicate that for a decay time of one year the peak cladding temperature can be limited to about 600°C. Figure 16 shows little difference between a full core and partial core discharge in this regard. Table VI indicates more precisely that the minimum decay time is about 280 days. Without any ventilation, however, NUREG/CR-0649 (Figure 21) concludes that peak cladding temperatures would reach 900°C within 20 hours if the loss of coolant occurs just 30 days after reactor offload. It also concludes that a minimum decay time of 3 years would still provide sufficient cooling for a 13.25-inch center to center square high-density storage rack (see Figure 24, no ventilation case), just by the room heat sinks and rejection through the walls and thermal radiation to the outside. An estimate of the amount of heat removal by these passive heat removal mechanisms was not provided in NUREG/CR-0649.
2. NUREG/CR-4982 (USNRC,1987) - NUREG/CR-4982 (USNRC,1987) also notes that for a PWR with high density fuel racks and a DCPP baseplate cooling hole diameter of 5 inches, the minimum decay time is also about 700 days if one assumes

perfect ventilation (see Table 3.1 and Section 3.1.3.1). The USNRC also notes that cladding failures, without sustained cladding oxidation, may occur for cooling times of up to 20% longer; i.e., 840 days. NUREG/CR-4982, used a software tool named SFUEL1W to determine the potential for cladding oxidation propagation. They concluded it would propagate to adjacent fuel assemblies that have decayed or cooled, for less than 2 years but not for 3 years (see Table 3.2). They also concluded that propagation would not occur to spent fuel that has decayed for three years or longer, even without ventilation (see Section 3.1.3.2). Note that the statement in Section 4.2 that, "If the discharged fuel is put into high density racks the air cooling capability is limited such that the critical cooling time is increased to one to three years and the conditional probability of a Zircaloy fire is increased to nearly 1.0", does not strictly apply to DCPP whose refueling cycles are longer than one year. In this older analysis, the conditional fuel fire probability was averaged over a one-year cycle whereas DCPP has a refueling cycle of about 1.5 years.

3. NUREG-1353 (USNRC, 1989) - In NUREG-1353 (USNRC, 1989), for a PWR with high density fuel racks and 5-inch diameter baseplate coolant hole size, the minimum decay heat time of 700 days is quoted from the older NUREG/CR-0649 report. The critical decay time is noted as about 700 days (Table 4.5.1). DCPP racks at 10.93 inches pitch are considered medium density racks (i.e., pitch of 9 to 13 inches) one place in one part of this report. However, the same report defines high density PWR configurations in another place as those having center-to-center spacing of 10.25 inches (Page 4-9). Zr-air oxidation event propagation to adjacent assemblies is stated to be unlikely if the older fuel temperatures are not already heated to within at least 100 to 200°C of the self- sustaining oxidation temperature; i.e., of 900°C (see page 4-9). This observation is also cited from NUREG/CR-4982. It is further stated that only fuel stored less than 2 years could be propagated to and only under some conditions; i.e., it would have to be next to the hottest assemblies (Page 68). This report also concludes that additional propagation to even lower power assemblies is highly unlikely. However, these calculations were not confirmed in the Sandia integrated fuel assembly heat up experiments which showed zero power adjacent fuel assemblies could still be susceptible to fuel fire propagation to from a fuel assembly with a power level of 15 kW.
4. NUREG/CR-6451 (USNRC, 1997) - In NUREG/CR-6451 (USNRC, 1997) for PWR high density rack geometries with a 5-inch baseplate coolant hole diameter, it is noted that in NUREG/CR-4982 that 700 days was chosen as the minimum decay time to avoid sustained cladding oxidation. For NUREG-6451, BNL developed a new software tool referred to as SHARP (Spent-Fuel Heatup: Analytical Response Program) (USNRC, 2002). Figure 6.4 of the SHARP manual indicates that for a high density PWR fuel assembly with a 5-inch baseplate coolant hole size (such as for DCPP), 60 GWD/MTU burnup, that a 5 volume per hour building ventilation rate can limit peak cladding temperatures to less than 800°C after 5 years of decay time in the SFP. Figure 3.1 of NUREG/CR- 6451 indicates the minimum decay times for high density PWR racks as a function of maximum cladding temperature and assuming a bounding burnup of 60 GWD/MTU. Figure 3.1 only goes up to fuel assembly temperatures of 700°C for which it indicates a minimum decay time of just more than one year. The authors of NUREG/CR-6451 then chose to lower the critical temperature from 900°C to 565°C; i.e., where the latter was selected as an incipient temperature for cladding failure. Adopting this new limiting temperature yielded a shorter minimum decay time than in NUREG/CR-4982; i.e., 17 months, or about 517 days (see Section 3.1.3). The selection of 565°C as the limit was made for

other purposes and should not be considered as a revised limit for the onset of Zr-air self-sustaining oxidation.

5. NUREG-1738 (USNRC, 2001) - In NUREG-1738 (USNRC, 2001) it is asserted that for today's fuel assembly rack configurations, burnup, and enrichment, a zirconium cladding fire may occur if there is a los of coolant in the SFP following a cooling period as long as 5 years (i.e., after reactor shutdown (see page A2C-2). However, in Section 3.7.1 NUREG-1738 states that it is expected that the fire propagation to other fuel assemblies would be limited to less than two full cores even after just one year of cooling. The USNRC assumed a more bounding 3.5 reactor cores for their purposes of obtaining a bounding source term to use in their assessment of consequences from SFP Loss of coolant events. Appendix 1.B indicates that if the SFP loss of coolant occurs early, then air rather than steam is the dominant reactant with the Zr cladding and the onset of self-sustaining oxidation may occur at lower temperatures when the cladding balloons; i.e., at 600°C. However, the integrated fuel assembly tests (NUREG-, 2015) did not observe this lower temperature effect even though cladding ballooning was observed. Instead, in the integrated fuel assembly heatup tests, ignition occurred at 900°C. On the other hand, if steam rather than air is the dominant environment, as would be the case if the sequence involves a slow boil-off of coolant, then oxidation of the Zr cladding is judged more protective and 900°C is suggested as a better indicator for the onset of self-sustaining oxidation. This suggests that steam environments are more resistant to propagation to cooler fuel assemblies once the self-sustaining oxidation is initiated.
6. In COMSECY-13-0030 (USNRC, 2013), the analysts were faced with having to estimate whether following a seismic event with high accelerations that resulted in a total loss of SFP coolant or a partial spent fuel pool drain down, would be coolable. A conservative assumption of it not ever being coolable was assumed, see Table 41. At the same time, a 75% Cesium release fraction was assumed as a base case cesium release fraction for a group of NPPs like DCPP, see Table 52. The 75% value was obtained from NUREG-1465 (USNRC, 1995), which evaluated the source terms released following a severe accident involving a reactor core. The 75% figure is for release caused by a severe accident from the reactor coolant system to the containment; i.e., it was not for release to the environment. This 75% estimate for cesium release would appear to bound what might be expected for the SFP. Also in Table 52, a high cesium release fraction estimate of 90% was identified.
7. (PG&E, 2013) - PG&E has computed DCPP specific times for the SFP coolant to reach 200°F for sequences in which there is a loss of SFP cooling but not a leakage of coolant (PG&E, 2013). The times to reach 200°F are a function of the initial water temperature as well as the days after shutdown. For the 125°F initial temperature, the times to 200°F are less than 10 hours for approximately the first 50 days. By 100 days after shutdown, 15 hours are available, and within 1 year, 25 hours are available. These times are until the time when substantial evaporation from the SFP can be expected, thereby complicating local recovery actions. The times to fuel uncovery would be shorter if there is also SFP coolant leakage.
8. (EPRI, 2014) – EPRI has performed a review of numerous past documents (EPRI, 2014) and summarized observations for a PWR SFP following a loss of SFP coolant. Some of these observations are noted below:
   i) For SFP cooling in high density racks, the fuel assemblies are coolable by air after a discharge time of 250 days decay with 50% probability of success if the coolant level is below the bottom of the fuel racks permitting air-cooling by natural circulation (See Sections D.2.1.2 and F.2.4). Section F.2.4 references tables developed from MAAP runs but these could

not be located.

ii) Both MELCOR and MAAP runs indicate that for a BWR, if the fuel assemblies each have power levels less than 3 kw, then they are coolable in air without resulting in Zr-air oxidation. They would be coolable at less than 2.5 kw even for a uniform configuration of assemblies, but would not be coolable if the heat load is greater than 4.75 kw for a uniform configuration of assemblies. The EPRI report argues that PWR fuel assemblies would be similar. (See Pages E-5 and E-8).

iii) A Zircaloy fire in the SFP may only occur due to a significant loss of coolant event such that the SFP is drained below the bottom of the fuel racks prior to fuel heatup and melt. In addition, for slow SFP boil down events, fuel melt is expected to occur prior to potential evaporation of the SFP inventory. For these cases, the fuel is judged to relocate such that inadequate air flow through the fuel racks would make the probability of a Zircaloy fire very low. Zircaloy fires could therefore only occur for larger loss of coolant events. Zircaloy fires would not apply to SFP boil down type events or to smaller losses of coolant (See Section F.2.4).

iv) In the MAAP runs performed, the aged fuel assemblies surrounding the hottest fuel assemblies modeled in a 1x4 arrangement were selected to have decayed for more than 69 months; i.e., almost 6 years. If the PWR fuel cells do not have the structural integrity to withstand high temperatures during a severe accident it is stated: "Therefore, the PWR fuel (depending on the design and the severe accident progression) may allow air ingress to the fuel assembly from elevations above the bottom of the fuel. The ability to cool the fuel rods is not clear" (Page E-5). The idea is that once the fuel cells fail, there could develop air cross flow between fuel cells (typically above the core midplane as coolant levels drop), but this would occur only after the fuel cell cans have failed structurally.

9. (Sandia, 2015 and USNRC, 2016a) - Integrated heatup tests performed at Sandia (Sandia, 2015 and USNRC, 2016a) for loss of coolant accidents in a SFP with high density PWR fuel assemblies, concluded that air-cooling by natural convection would limit cladding temperatures to less than the Zr-air ignition temperature (i.e., about 900°C) if the fuel assembly power is less than 8 kw (i.e., for a nominal fuel assembly with 45 GWD/MTHM burnup after 8 months decay). Sandia also concluded that for fuel assemblies with decay power of 15 kw (i.e., for a nominal fuel assembly with 45 GWD/MTHM burnup after 86 days decay), that the fuel assembly heatup would exceed 900°C and ignite. Furthermore, it was concluded that the centrally heated 15 kw fuel assembly would propagate to the surrounding, cooler fuel assemblies in a 1x4 arrangement causing them to also overheat. The integrated test configuration was limited to just 5 fuel assemblies. The basis for the air-cooling flow rates to each fuel assembly was not fully described in the SANDIA reference. The cooling times for an average fuel assembly at DCPP to reach these heat loads are consistent with those listed in Table 8-7 of the current assessment.

10. (USNRC, 2016) – The USNRC (USNRC, 2016) has performed heat up calculations to determine the time available to respond to an SFP instantaneous loss of coolant accident. The time is measured from the total loss of coolant until the fuel cladding heats up to 900°C. The results were scaled from a MELCOR analysis for BWR SFPs to apply to PWR high density SFPs. For 60 GWD/MTU burnups, Figure 8 shows a 0.1 years (36 days) of decay time allows 2 hours, 0.3 years decay (i.e., 3 months) allows 3 hours, 1 year decay allows 7 hours, 1.64 years allows 10 hours, 2 years allows 12 hours, and 3 years allows 18 hours until the hottest fuel assembly reaches 900°C in an adiabatic calculation. A moderate loss of coolant rate (6-inch

diameter hole size) can add 2.5 hours to the total time available and a small loss of coolant rate (1.5-inch diameter) can add 24 hours.

11. (PG&E, 2018, Table 9.1-1) - Per the DCPP UFSAR (PG&E, 2018a, Table 9.1-1) the total SFP heat load at 100 hours after a full reactor core offload (i.e., 193 fuel assemblies) is conservatively estimated as 10.74 MW (i.e., 36.67E+6 BTU per hour). The results in Table 8-5 developed for the current assessment are slightly higher than the DCPP UFSAR 100-hour heat load for both typical refueling outages and for the EOL shutdown offload. During the first 30 days after reactor shutdown, the evaluated SFP heat load presented in Table 8-5 for the full core offload exceeds 5.34 MW. A 5.34 MW power level or greater is estimated for a generic PWR SFP (Fauske, 2019) to result in SFP boiling rather than have temperatures level out at 200°F if all SFP cooling was lost; i.e., it would require SFP pool boiling, rather than just an increased coolant evaporation rate, to limit the SFP temperature increase. However, for a typical refueling, after the reactor is refueled, or after 30 days, the total DCPP SFP heat load would be less than the 5.34 MW limit, and so evaporation would be sufficient for DCPP to limit further SFP temperature rises. For the EOL offload, all fuel remains in the SFP. Per the assessed total SFP heat load in Table 8-5 of the current assessment, the total SFP decay heat is less than the 5.34 MW limit mentioned within 60 days.

12. (Fauske and Associates, 2019) - In 2019, a paper describing the application of MAAP 5.02 to analysis of a loss of SFP cooling for a typical PWR, but with no SFP coolant leakage was published (Fauske and Associates, 2019). The reactor core for the plant analyzed had a reactor core consisting of 157 fuel assemblies; i.e., it is smaller than the DCPP reactor cores. Key points from the paper's base case analysis are summarized below:
    a. The loss of SFP cooling was simulated as initiating 7 days after reactor shutdown when the total SFP decay heat was at 9.15 MW. Coolant boiling was found to occur after 10 hours. For an accident start time of 4-days after reactor shutdown used as sensitivity analysis, the total decay heat in the SFP was initially 12.74 MW. These total SFP heat loads are slightly less than what is described in Table 8-7 of the current assessment for DCPP. A conservatism in these MAAP runs is that the SFP heat load was held constant after loss of SP cooling initiation rather than decayed with time over the several days duration that was simulated following the loss of SFP cooling.
    b. Without actions for SFP makeup and restoration of cooling, coolant boil-off was evaluated to reach the Top of the Active Fuel (TAF) after 75 Hours, and to reach fuel mid-level after 91 Hours. Zr-steam oxidation then was found to begin generating hydrogen, and cesium release occurs at 93 hours.
    c. Water Level reached Bottom of Active Fuel (BAF) at 140 Hours, allowing air natural circulation through the racks to begin only after 149 hours.
    d. The hottest fuel racks were found to collapse at 153 Hours, while no other racks collapsed. Once collapsed, the corium temperatures at the bottom of the SFP were found too low to sustain MCCI. The releases of cesium also ended at 153 hours.
    e. A hydrogen explosion occurred in the FHB at 163 Hours, and all water and steam in the FHB were then depleted after another 5 hours.

f. At 336 hours, all Zr-Air oxidation within the SFP ceased.
g. Figure 15 of the paper indicates that after 153 hours only the fuel in racks 1 and 2 collapsed (i.e., the ones with the latest full core offload), whereas the fuel in racks 3, 4, 12, and 13 (i.e., those placed close to the hottest fuel and with relatively high stored power) did not collapse but exhibited blockage and experienced temperatures of about 1200°C; i.e., not yet above the 2300°C threshold temperature for Cesium release. The 1x4 arrangement of fuel assemblies was not included in the analysis case. The fuel in the remainder of the 15 racks did not reach temperatures greater than 800°C. A separate case with the fuel arranged in a 1x4 arrangement made little difference to the relocated mass of fuel.
h. Figure 44 of the paper describes a sensitivity case (i.e., 9) which is similar in specification to case 1 described above but with 8 added nodes representing the FHB free volume. The 8-node FHB model sequesters condensation on the walls of the FHB limiting the effectiveness of convective cooling and blocking natural convection for most of the transient. For case 9 almost the entire fuel mass was found to relocate onto the SFP floor. What doesn't relocate in racks 14 and 15 amounts to only a couple of dozen fuel assemblies; i.e., these racks were modeled as nearly empty and likely have individual fuel assembly power levels of just 0.7 kw per assembly. Fuel assemblies at DCPP that have decayed for 6 years would have power levels of approximately 0.7 kW.
i. The MAAP models for the PWR SFP design considered did not account for the axial flux trap (gap) design which provides an extra channel for coolant flow. This design is implemented at DCPP.