SAFETY EVALUATION REPORT

Docket No. 72-26
Pacific Gas & Electric Company
Materials License No. SNM-2511
Amendment No. 3

1 SUMMARY

By letter dated July 31, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML122270603), as supplemented March 14 (ML13086A113), May 23, (ML13175A184), and September 5, 2013, (ML13259A296), Pacific Gas and Electric Company (PG&E) submitted license amendment request (LAR) 12-003 to the United States Nuclear Regulatory Commission (NRC) to amend Materials License No. SNM-2511 for the Diablo Canyon (DC) specific independent spent fuel storage installation (ISFSI) in accordance with 10 CFR Part 72. The DC ISFSI utilizes a Holtec HI-STORM 100 Cask System modified for its specific site design and safety requirements, which employ multi-purpose canisters (MPC), placed inside concrete and steel overpacks to safely store the spent nuclear fuel (SNF). The amendment increases the allowable heat load, clarifies how to calculate heat load for regionalized loading of high burn-up fuel (HBF), revises the helium backfill range for certain MPCs, provided clarification that the use of the supplemental cooling system is only applicable to SNF previously transferred, and revises the DC ISFSI maximum average storage and maximum transfer temperatures to reflect more accurate site data.

LAR 12-003 proposed the following changes to the technical specifications (TS):

1. Revise tables 2.1-7, 2.1-8, and 2.1-9 in TS 2.0, "Approved Contents," to allow up to a 28.74kW heat load for uniform loading and 25.572kW heat load for regionalized loading. This changes the maximum allowable decay heat per storage location, in watts, determined from Table 2.1-7 or 2.1-9 to be consistent with this proposed license amendment request. Revise Table 2.1-8 to delete the note that limits Westinghouse Zirconium alloy low oxidation (Zirlo™) clad fuel to a burnup of 45,000 MWD/MTU and replace the existing Note 3 with a note that refers to TS 2.3, "Alternate MPC-32 Fuel Selection Criteria."

2. Revise TS 2.3, "Alternate MPC-32 Fuel Selection Criteria," to add reference to Table 2.1-9 as regionalized loading of HBF.

3. Revise TS 3.1.1, "Multi-Purpose Canister (MPC)," Surveillance Requirement (SR) 3.1.1.2 to add a new helium backfill pressure range for MPCs with heat loads less than or equal to 28.74kW.

4. Revise TS 3.1.4, "Supplemental Cooling System," applicability to only be applicable for unloading of high burnup fuel loaded in 2012 under the provisions of License Amendment No. 2.

5. Add TS 4.1.3 to Design Features Important to Thermal Analysis-
   a. Providing for a maximum average yearly temperature of 65° as the basis for a loaded overpack in the cask transfer facility, or storage on the ISFSI pad.
   b. Providing for a maximum temperature of 100°F, averaged over a 3-day period, as the basis for transfer activities in the transfer cask.

1 BACKGROUND

The DC ISFSI is co-located with the Diablo Canyon Power Plant (DCPP) on PG&E-owned property, located on the California coast approximately 10 km [6 mi] northwest of Avila Beach, California. The DCPP consists of two nuclear-generating units, each having a spent fuel pool to store spent nuclear fuel generated from reactor operation. The DC ISFSI has provided additional spent nuclear fuel storage capacity since 2006.

The DC ISFSI is designed to hold up to 140 storage casks. The DC ISFSI consists of a modified Holtec HI-STORM 100 Cask System, a cask transfer facility (CTF), an onsite cask transporter, and the cask storage pads. In addition, to accommodate spent nuclear fuel generated during the licensed period of the ISFSI, as well as any damaged fuel assemblies, debris, and nonfuel hardware, license no. SNM-2511 authorizes the use of other MPC designs from the HI-STORM 100 Cask System, including the MPC-24, MPC-24E, and MPC-24EF.

The primary intent of LAR 12-3 is to provide an updated thermal analysis that determined the DC ISFSI can safely store SNF with a higher heat load. The licensee utilized this analysis to determine that acceptable thermal margin remained to prevent exceeding SNF thermal limits during onsite transfer without the use of supplemental cooling.

2 REVIEW CRITERIA

The staff evaluation of the proposed changes is based on ensuring PG&E continues to meet the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. The staff used the guidance in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities” during its evaluation. The staff evaluation focused only on changes to SNM-2511 and associated TS that were requested in the application. The staff did not reassess previously approved areas of the license, TS, and the FSAR not affected by the proposed changes or those areas of the FSAR modified by PG&E as allowed by 10 CFR 72.48. The technical objectives for the following review disciplines are described below for each of the proposed changes.

3 OPERATIONAL SYSTEMS

The licensee did not propose any changes that affect the operational system evaluation provided in the safety evaluation reports (SERs) supporting the original license issued on March 22, 2004, Amendment No. 1 issued February 10, 2010, or Amendment No. 2 issued January 19, 2012. Therefore, the staff determined that a new operational system evaluation was not required.

4 SYSTEM, STRUCTURE, AND COMPONENT (SSC) AND DESIGN CRITERIA EVALUATION

The licensee did not propose any changes that affect the staff's SSC and design criteria evaluation provided in the safety evaluation reports (SERs) supporting the original license issued on March 22, 2004, Amendment No. 1 issued February 10, 2010, or Amendment No. 2 issued January 19, 2012. Therefore, the staff determined that a new SSC and design criteria evaluation was not required.

5 STRUCTURAL AND MATERIALS EVALUATION

The TS changes have been evaluated by the staff with respect to their potential to affect the integrity of cask materials and structural performance as required by 10 CFR 72.24 (a) ,(b), (c)(3)10, CFR 72.82 (c)(2), and 10 CFR 72.106 (a), (b), and (c).2. By increasing the allowable heat load, the licensee is requesting a reduced cooling time for spent nuclear fuel at a particular burn-up. The maximum allowable burn-up is not changed and remains limited to 68,200 MWD/MTU currently specified in the TS. The structural and materials evaluation focused on ensuring that the temperature of SSCs important to safety (ITS) could not increase beyond their design allowable temperature limits given the increased heat load. The potential for increases in temperature resulting from these changes has been analyzed by the licensee using thermal analysis methods and material properties previously evaluated in the DC ISFSI FSAR. Under the proposed amendment, the cask SSCs remain within their previously approved design allowable temperature limits.

The proposed amendment would also result in an increase in internal pressure. This change could result in increased loading on SSCs ITS beyond their design allowable. However, while this proposed change increases the required backfill pressure, the licensee's analysis, evaluated and confirmed by the NRC as discussed below, determined that the pressure remains within the parameters of the previously approved design.

The staff evaluation of the licensee's analyses included confirmation of input values used in the licensee calculation packages, along with a review of design details used to provide parameters in analyses. The staff determined and confirmed that the proper material properties and boundary conditions were used. The staff determined that the licensee's selected models accurately reflected the specific design parameters, and that the assumptions and modeling parameters were consistent with the review guidelines in Section 6.4.4 of NUREG-1567, and that the licensee's assumptions were adequate for the structural and materials characteristics in the HI-STORM 100 shortened/anchored (SA) geometry and analyzed conditions. The licensee's engineering drawings were reviewed to verify that adequate geometry dimensions were translated to the analysis models. The material properties presented in the FSAR were reviewed to verify that they were appropriately referenced and used. Finally, the staff determined that the licensee-provided proposed updated FSAR sections included accurate information that allowed the staff to make a safety determination on the acceptability of the proposed amendment to the DC ISFSI.

5.1 Evaluation Findings

F5.1 The FSAR and docketed materials relating to the description of confinement structures, systems and components would meet the requirements of 10 CFR 72.24 (a) and (b), 10 CFR 72.82 (c)(2), and 10 CFR 72.106 (a), (b), and (c).2 under the proposed amendment.

F5. The FSAR and docketed materials relating to suitable material properties for use in the design and construction of the SSCs would meet the requirements of 10 CFR 72.24(c)(3) under the proposed amendment.

6.0 THERMAL EVALUATION

6.1 Review Objective

The objective of the LAR 12-03 DC ISFSI thermal evaluation is to confirm that the decay heat removal system is capable of reliable operation given the proposed increased heat load. The review also evaluates whether the temperatures of materials using SSCs ITS and fuel assembly cladding material remain within the allowable limits under normal, off-normal, and accident conditions.

The following proposed revisions to the TS are applicable to the thermal evaluation.

- Tables 2.1-7, 2.1-8, and 2.1-9 in TS 2.0, "Approved Contents," would allow up to a 28.74kW heat load for uniform loading and 25.572kW heat load for regionalized loading. This would change the maximum allowable decay heat per storage location, in watts, determined from Table 2.1-7 or 2.1-9 to be consistent with this proposed license amendment request. Table 2.1-8 would be revised to delete the note that limits Zirlo clad fuel to a burnup of 45,000 MWD/MTU and replace the existing Note 3 with a note that refers to TS 2.3, "Alternate MPC-32 Fuel Selection Criteria."

- TS 2.3, "Alternate MPC-32 Fuel Selection Criteria," would be revised to add reference to Table 2.1-9 as regionalized loading of high burn-up fuel (HBF).

- TS 3.1.1, "Multi-Purpose Canister (MPC)," Surveillance Requirement (SR) 3.1.1.2 would be revised to add a new helium backfill pressure range for MPCs with heat loads less than or equal to 28.74kW.

- TS 3.1.4, "Supplemental Cooling System," Applicability would be changed to only be applicable for unloading of high burnup fuel loaded in 2012 under the provisions of License Amendment 2.

- TS 4.1.3 Design Features Important to Thermal Analysis:
  a. A maximum average yearly temperature of 65°F would be the basis for a loaded overpack in the cask transfer facility, or storage on the ISFSI pad.
  b. A maximum temperature of 100°F, averaged over a 3-day period, would be the basis for transfer activities in the transfer cask.

6.2 Evaluation

6.2.1 Licensee Thermal Models

The licensee performed a site-specific thermal analysis to verify that the DC ISFSI MPC-32 design remained in compliance with the limits previously approved by the staff. The licensee's analysis concluded that for all conditions of system operation with a design basis heat load of 28.74 kW for uniform heat loading and 25.572 kW for regionalized loading, the temperature limits provided in NUREG -1567 and Interim Staff Guidance (ISG) SFST-11, Rev. 3 were met. In LAR 11-001, the licensee updated the DC ISFSI thermal analysis to a three-dimensional (3-

D) computational fluid dynamics (CFD) analysis. In LAR 12-03 the analysis was modified to address the storage of high burnup fuel consistent with the evaluation guidance of ISG-SFST-11, Rev. 3. The licensee's analysis concluded that fuel cladding temperatures were maintained below the NUREG-1567 limits for the specified operating conditions. The licensee determined that the supplemental cooling system is only applicable for unloading of high burnup fuel loaded in 2012 under the provisions of License Amendment No. 2. The licensee's developed 3-D FLUENT CFD model is described below:

The interior of the MPC is a 3-D array of square shaped cells inside an irregularly shaped basket outline confined inside the cylindrical space of the MPC cavity. The fuel bundles inside the fuel cell for the pressurized water reactor fuel assemblies are replaced by an equivalent porous medium using the flow impedance properties determined using a pressurized water reactor (PWR) 17X17 fuel assembly thermal-hydraulic characterization test performed at Sandia National Laboratory. The equivalent effective thermal properties of the porous medium are the same as that used in the HI-STORM 100 Cask System, CoC No.1014. The internal components of the MPC cavity, including the basket cross-section, bottom flow holes and plenums are modeled individually. The stainless steel plates in the MPC basket wall have Metamic panels and sheathing attached. The arrangement of metal layers results in the composite wall having different thermal conductivities in the in-plane (parallel to panel) and out-of-plane (perpendicular to panel) directions. The effective thermal properties of the basket sandwich are consistent with the values previously used in the licensee's thermal evaluations supporting the HI-STORM 100 Cask System. The overpack shells, lid and the radial concrete shield are modeled explicitly. The plates on the top of the lid and the bottom of the overpack are modeled as conducting walls, thereby including the thermal conduction resistance along these plates in the model. The inlet and outlet vents in the HI-STORM 100 shortened/anchored (SA) overpack are modeled explicitly. The model used by the licensee includes three modes of heat transfer - conduction, convection and radiation. The helium flow within the MPC is modeled as laminar. Surface to surface thermal radiation heat transfer is modeled using the discrete ordinates (DO) radiation model in the FLUENT model. The airflow through the annular space between the MPC and the overpack is modeled as transitional turbulent flow using k-ω turbulence model to incorporate the effect of air turbulence on the systems thermal performance. Insolation on the outer surface of HI-STORM 100SA overpack is based on the 12-hour levels averaged on a 24-hours basis.

The cask transfer facility (CTF) is a steel cylinder backed by concrete. The licensee's model assumed that the CTF cylinder is a perfect insulator that does not permit heat from the HI-STORM 100SA to be absorbed by the CTF structure and the surrounding soil. The licensee stated this is a conservative assumption because it maximizes the computed temperatures of both the HI-STORM 100SA and the CTF.

The licensee's 3-D model implemented to analyze the HI-TRAC has the following characteristics: The MPC portion of the model contains a porous medium to represent the fuel, the top and bottom plenum, and a fluid (helium) zone in the basket-to-shell downcomer region. Radiation heat transfer between the periphery of the fuel basket and the inner surface of the MPC shell is included using FLUENT's DO radiation model. In the radial direction, the HI-TRAC portion of the model explicitly contains five layered zones that represent the inner shell, the radial lead shield, the outer shell, the water jacket and the enclosure shell. The water density in the water jacket is specified as a function of temperature and the annular gap between the MPC and HI-TRAC is modeled individually as a fluid (air) zone.

The staff reviewed the applicant's description of the DC ISFSI MPC-32 thermal model. Based on the information provided in the application regarding the thermal model, the staff determines that the application is consistent with guidance provided in Section 6.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1567 which provides that the applicant should present a thermal analysis that clearly demonstrates the storage system's ability to manage design heat loads and have the various materials and components remain within temperature limits. Here, the applicant has provided a detailed description of thermal models used to perform the evaluation of the storage cask with sufficient detail for the staff to make a determination of the adequacy of the models to represent the heat transfer characteristics and environment prevailing at the storage site. Therefore, the staff concludes that the description of the thermal models acceptable because the description satisfies NUREG-1567 and the requirements in 10 CFR 72.122 and 72.128.

6.2.2 Licensee Analyzed Configurations

The licensee used the above thermal models to perform the thermal evaluation for normal conditions of transfer, transfer of the MPC to the overpack in the CTF configuration, and long term storage at the ISFSI pad. The licensee considered two scenarios that are described as Scenario 1 and Scenario 2 as follows:

1) Scenario 1 with a total head load of 28.74 kW, uniform loading, and MPC absolute operating pressure of 557287.5 Pa.
2) Scenario 2 with a total head load of 25.572 kW, regionalized loading, and MPC absolute operating pressure of 537022.5 Pa.

The licensee requested a total heat load limit of 28.74 kW for the MPC-32 loaded with high burnup fuel. Scenario 1 was evaluated in the application because the licensee concluded that the total heat load and pressure bounded Scenario 2 in terms of fuel cladding temperatures and cavity pressures. The CTF configuration was evaluated based on Scenario 1. This configuration consists of a loaded HI-STORM 100SA overpack that cannot be removed from the CTF because of a failure of the equipment that lifts the HI-STORM 100SA. Under such a condition, the flow of air to the bottom inlet vents would be restricted. The licensee's approach to evaluate the CTF was by a steady state analysis that evaluated the peak cladding temperature remained below allowable limits.

The staff reviewed the analysis configuration and approach to find the bounding configuration (which the licensee concluded to be Scenario 1), and the staff determined this scenario would result in maximum peak cladding temperatures as compared to Scenario 2. The staff reviewed the analysis approach used to perform the thermal evaluation in the CTF and determined that a steady state analysis while in the CTF configuration would result in maximum cladding temperatures as compared to a transient analysis (licensee scenario 1). The staff also verified the analysis itself to determine that the peak cladding temperatures predicted by the applicant would remain below ISG-11 Rev. 3 recommended limit of 400°C during normal storage and short-term operations. Based on the information provided in the application regarding the description of the analyzed configurations, the staff determines that the application is consistent with guidance provided in Section 6.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1567, which provides that the licensee should present a thermal analysis that demonstrates the ability to manage design basis heat loads and have the various materials remain within temperature limits. Here, the licensee has demonstrated this ability by accurately identifying the bounding scenario, and demonstrating that the analysis result from the bounding scenario is lower than the recommended limit of 400°C. The available margin includes the

discretization error, as described in Section 6.2.2.2 Therefore, the staff concludes that the description of the analyzed configurations is acceptable because the description and analyzed configurations satisfy NUREG-1567 and the requirements in 10 CFR 72.122 and 72.128.

6.2.2.1 Licensee's Thermal Evaluation for Normal Storage

The licensee's evaluation was performed to support the DC ISFSI license for a maximum uniform heat load of 28.74 kW, a helium backfill pressure of 34 psig (234.42 kPa) at 70°F, and an MPC operating pressure of 5.5 atmospheres (80.83 psig). The normal long-term storage condition of the HI-STORM 100SA overpack on the ISFSI pad is bounded by the HI-STORM 100SA overpack in the CTF configuration since the flow of air to the bottom inlet vents would be restricted in the CTF.

The licensee performed a grid convergence study to obtain the discretization error for the storage configuration. The discretization error is determined using the procedure specified in American Society of Mechanical Engineers, "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," (ASME V&V 20-2009, November 30, 2009) which is the accepted industry guidance for this type of analysis.

The staff reviewed the analysis configuration and approach to find the bounding configuration (which the licensee concluded to be the HI-STORM 100SA overpack in the CTF) and determined this configuration would result in maximum cladding temperatures as compared to the normal storage condition on the ISFSI pad. The staff reviewed the analysis approach used to perform the thermal evaluation on the ISFSI pad and determined that a steady state analysis would result in a larger margin as compared to the CTF configuration. See Section 6.2.2 for additional details on the staff's review.

6.2.2.2 Licensee's Thermal Evaluation for the HI-STORM 100SA in the Cask Transfer Facility

The off-normal scenario the licensee analyzed is a loaded HI-STORM 100SA overpack that cannot be removed from the CTF because of a failure of the equipment that lifts the HI-STORM 100SA. Under such a condition, the flow of air to the bottom inlet vents would be restricted. The licensee performed a steady state calculation for this condition using the 3-D FLUENT CFD model for the DC ISFSI heat load described earlier. The analysis results show the fuel cladding temperature and other MPC and overpack temperatures remained below their respective long-term normal operating temperature limits. Also, the results showed the licensee's calculated maximum operating pressure remained below the normal design pressure limit as documented in the DC ISFSI FSAR. The licensee's analysis concluded that the HI-STORM 100SA overpack can be loaded at the CTF for an indefinite time for the DC design basis maximum heat load of up to 28.74 kW.

The licensee performed a grid convergence study to obtain the discretization error for the CTF configuration. The discretization error is determined using the procedure specified in American Society of Mechanical Engineers, "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," (ASME V&V 20-2009, November 30, 2009) which is the accepted industry guidance for this type of analysis.

The staff reviewed the analysis configuration and approach to evaluate the off-normal scenario of the loaded overpack that cannot be removed from the CTF due to equipment malfunction. The staff reviewed the licensee's approach to perform steady state analysis for this

configuration and determined the peak cladding temperatures were below the recommended limit, as concluded by the licensee. See Section 6.2.2 for additional details on the staff's evaluation.

6.2.2.3 Licensee's Thermal Evaluation During Transfer Operations

The licensee performed calculations to evaluate the temperature and pressure fields in the HI-TRAC loaded with an MPC-32 in a vertical (upright) orientation. The licensee determined that Scenario 1 is the limiting fuel storage configuration during on-site transfer operations as it provides the highest heat load and pressure. The licensee's results show that the peak fuel cladding temperature during normal on-site transfer conditions remained below its temperature limit for both moderate burnup fuel and high-burnup fuel. Also, the results of the licensee's analysis indicated the maximum operating pressure remains below the normal design pressure limit in this configuration.

The licensee performed a grid convergence study to obtain the discretization error for the transfer configuration. The discretization error is determined using the procedure specified in American Society of Mechanical Engineers (ASME), "Standard for Verification and Validation in

Computational Fluid Dynamics and Heat Transfer," (ASME V&V 20-2009, November 30, 2009) which is the accepted industry guidance for this type of analysis.

The staff reviewed the licensee's thermal evaluation during transfer operations and verified the maximum cladding temperatures predicted by the applicant would remain below ISG-11 Rev. 3 recommended limit of 400°C for high burnup fuel and 570°C for low burnup fuel. Based on the information provided in the application regarding the transfer operations, the staff determines that the application is consistent with guidance provided in Section 6.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1567 which provides that the licensee should present a thermal analysis that demonstrates the ability to manage design basis heat loads and have the various materials remain within temperature limits. Here, the licensee has demonstrated this ability by performing the calculation and demonstrating that the analysis result is lower than the recommended limit of 400°C for low burnup fuel and 570°C for high burnup fuel. The available margin includes the discretization error, as described in Section 6.2.2.2. Therefore, the staff concludes that the evaluation during transfer operations is acceptable because the evaluation and analysis satisfy NUREG-1567 and the requirements in 10 CFR 72.122 and 72.128.

6.2.2.4 Licensee's Thermal Evaluation During Off-Normal and Accident Conditions

The licensee considered three off-normal conditions to be evaluated for the DC ISFSI storage configuration: off-normal ambient temperature, off-normal pressure, and partial blockage of air inlets. The licensee's analysis states that all the MPC and HI-STORM 100SA component temperatures remain below their temperature limits for the considered off-normal conditions.

The licensee considered four accident conditions for the DC ISFSI storage configuration: fire, 100% blockage of inlet ducts, extreme ambient temperature, and burial under debris. For the fire accident event, the licensee's analysis demonstrated that the fuel temperature rise is small and all MPC and overpack components temperatures remain below temperature limits. Since the temperature increase is small in the MPC, the pressure increase is small compared to normal storage. For the 100% blockage of inlet ducts event, the licensee concluded that for a blockage duration of 32 hours, all MPC and overpack components temperatures remain below temperature limits. Also, the calculated pressure remained below the accident limit during this

event. For the extreme ambient temperature event, the licensee assumed that an extreme temperature of 125°F persists for a sufficient duration to reach steady state conditions. From the steady state analysis, the licensee demonstrated that the calculated temperatures remained below accident limit guidance of 570°C (1078°F) provided in NUREG 1567.

The licensee considered three accident events during transfer: water loss accident condition, fire accident, and tornado missile impact. For the water loss accident event, the licensee's analysis shows that the peak fuel cladding temperature remains below its temperature limit and all the MPC and HI-TRAC overpack component temperatures remain also below their respective temperature limits. The pressure analysis also shows that the maximum pressure remains below its accident limit. For the fire event, the licensee performed the analysis to determine the duration and effects of an assumed 50-gallon flammable liquid fuel fire on the HI-TRAC transfer cask. The analysis showed little effect of an external fire event on MPC fuel temperatures as the large thermal capacity of the HI-TRAC transfer cask and its contents will continue to suppress rapid temperature changes. The calculation shows that fuel cladding and all component temperatures are below their accident temperature limits. The calculation also shows the calculated pressure remains well below the accident limit. For the tornado missile impact, the licensee stated that from thermal-hydraulic performance perspective this event is identical to the water jacket loss accident condition and is therefore bounded by that evaluation. The large thermal capacity of the HI-TRAC transfer cask and its contents will suppress rapid temperature changes.

The staff reviewed the licensee's thermal evaluation during off-normal and accident conditions and verified the maximum cladding temperatures predicted by the applicant would remain below ISG-11 Rev. 3 recommended limit of 570°C for all postulated off-normal and accident events. Based on the information provided in the application regarding off-normal and accident events, the staff determined that the application is consistent with guidance provided in Section 6.4.4 (Analytical Methods, Models, and Calculations) of NUREG-1567 which provides that the licensee should present a thermal analysis that demonstrates the ability to manage design basis heat loads and have the various materials remain within temperature limits. Here, the licensee has demonstrated this ability by performing the calculations and demonstrating that the analysis results are lower than the recommended limit of 570°C. Therefore, the staff concludes that the evaluation during off-normal and accident events is acceptable because the evaluation and analysis satisfy NUREG-1567 and the requirements in 10 CFR 72.122 and 72.128.

6.3 Staff Confirmatory Analyses and Conclusions

The staff evaluation of the licensee's thermal models to support the analyses included staff confirmatory analyses, confirmation of code input values used in the licensee calculation packages, along with a review of design details used to provide parameters in the computer models. The staff determined and confirmed that the proper material properties and boundary conditions were used. The staff determined that the licensee's selected code models accurately reflected the specific design parameters, and that the assumptions and modeling parameters were consistent with review guidelines in NUREG-1567. The staff determined that licensee assumptions were adequate for the flow and heat transfer characteristics prevailing in the HI-STORM 100SA geometry and analyzed conditions. The licensee's engineering drawings were reviewed to verify that adequate geometry dimensions were translated to the analysis models. The material properties presented in the FSAR were reviewed to verify that they were appropriately referenced and used. The staff determined that the licensee's sensitivity analysis calculations were correctly performed to obtain mesh-independent results that provided bounding predictions for all analyzed conditions during normal storage, transfer operations, and

off-normal and accident events. The staff determined that the licensee's utilization of computer modeling was consistent with the guidance of NUREG-1567 and ISG-21, "Use of Computational Software," 10 CFR 72.122, and 10 CFR 72.128. Finally, the staff determined that the licensee provided proposed updated FSAR sections that included accurate information that allowed the staff to make a safety determination on the acceptability of the proposed amendment to the DC ISFSI.

Therefore the staff finds the DC ISFSI thermal analysis and conclusions acceptable and that the DC ISFSI would continue to safely store spent nuclear fuel within TS parameters.

6.4 Evaluation Findings

F6.1 The submitted revised FSAR Chapter 2 describes SSCs ITS to enable an evaluation of their thermal effectiveness. SSCs important to safety continue to remain within their operating temperature ranges as required by 10 CFR 72.122.

F6.2 Under the proposed amendment, the DC ISFSI would continue to maintain heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask continues to provide adequate heat removal capacity without active cooling systems for steady state and transient operations as required by 10 CFR 72.128.

F6.3 Under the proposed amendment, the spent fuel cladding continues to be protected against degradation leading to gross ruptures under long-term storage by maintaining cladding temperatures below 752°F (400°C) in a helium environment as required by 10 CFR 72.122. Protection of the cladding against degradation continues to allow ready retrieval of spent fuel for further processing or disposal.

F6.4 Under the proposed amendment, the spent fuel cladding continues to be protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining cladding temperatures below the NUREG-1567 guidance temperature of 1058°F (570°C) in a helium environment maintaining 10 CFR 72.122 requirements. Protection of the cladding against degradation continues to allow ready retrieval of spent fuel for further processing or disposal.

F6.5 Under the proposed amendment, the staff finds that the thermal design of the DC ISFSI continues to remain in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria, as identified in NUREG-1567 and the DC ISFSI FSAR, would be satisfied. The evaluation of the thermal design provides reasonable assurance that under the proposed amendment, the cask continues to allow safe storage of spent fuel. This finding is reached based on a review that considered 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices as noted above.

7 SHIELDING AND RADIATION PROTECTION EVALUATION

The licensee evaluated the impact of the increased decay heat on the dose rates at the controlled area boundary and the radiation protection of the ISFSI operations in Appendix R of calculation HI-2002563, "Dose Evaluation for the ISFSI at Diablo Canyon Power Station," Revision 10 - Non-Proprietary Version and cover pages (ML13086A110). Based on a comparision of the source terms (both neutron and gamma) between the originally analyzed bounding source terms with that of the fuel to be loaded into the cask with 28.7 KW decay heat,

the licensee determined that the calculated dose rates of the cask with previously approved contents bound the dose rates of cask with the increased decay heat (determined by a combination of initial enrichment, burnup, and cooling time) and therefore the Radiation Protection analyses remain valid.

7.1 Evaluation Findings

The staff reviewed the licensee's analyses and determined that the applicant's assessment on cask radiation shielding and radiation protection are consistent with the review guidance of NUREG – 1567 and are acceptable and there is reasonable assurance that the DC ISFSI with 28.74 KW decay heat load casks continues to meet the regulatory requirements of 10 CFR Part 72.

Based on its review of LAR 12-03, the staff finds that SNM-2511, as amended, continues to meet the regulatory requirements of 10 CFR 72.104 and 72.106, and the acceptance criteria specified for both intact and damaged fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. In summary, the staff finds the following:

F7.1 SSCs important to safety with respect to radiation shielding are described in sufficient detail in LAR 12-03 to enable an evaluation of their effectiveness

F7.2 The staff finds that the criticality design features for SNM - 2511 and associated TS, as amended, continues to be in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied.

8 CRITICALITY EVALUATION

The staff reviewed and evaluated the impacts of the proposed revisions to criticality safety and found that the following proposed changes are relevant:

- Revised Tables 2.1-7, 2.1-8, and 2.1-9 in TS 2.0, "Approved Contents," to increase the allowable decay heat to 28.74 kW for uniform loading and 25.57 kW for regionalized loading.
- Revision of TS 2.3, "Alternate MPC-32 Fuel Selection Criteria," to add reference to the revised Table 2.1-9 for regionalized loading of high burnup fuel.

The staff evaluated the licensee's criticality safety analysis provided in LAR 12-03 and the licensee's supporting information to determine whether all credible normal, off-normal, and accident conditions were identified and their potential consequences on criticality safety considered such that SNM-2511, as amended, would meet the regulatory requirements of 10 CFR 72.24(c)(3), 72.24(d), and 72.124 under the proposed amendment.

The staff reviewed the technical basis provided in LAR 12-03. Based on its review, the staff determined that the proposed changes have no adverse impact on the criticality safety of the dry storage casks loaded with increased burnup and decay heat since the increased decay heat load has been determined to not impact the integrity of the fuel assembly and the structural integrity of the casks. The increased fuel burnup will not adversely impact the criticality safety of the casks because increased burnup will reduce the reactivity of the spent fuel, and therefore increase the criticality safety of the casks while increasing the heat load, as discussed above.

8.1 Evaluation Findings

Based on its review of LAR 12-03, the staff finds that SNM - 2511, as amended, would meet the regulatory requirements of 10 CFR 72.124 under the proposed amendment, for both intact and damaged fuel. This finding is reached based on a review that considered 10 CFR Part 72, , appropriate regulatory guides, applicable codes and standards, and accepted engineering practices as identified in the DC ISFSI FSAR. In addition, the staff finds the following:

F8.1 Under the proposed amendment, the DC ISFSI system would be subcritical under all credible conditions.

F8.2 The NRC staff finds that the criticality design features for SNM - 2511 and associated TS under the proposed amendment would operate in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied.

9 CONFINEMENT EVALUATION

The licensee did not propose any changes that affect the staff's confinement evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required.

10 CONDUCT OF OPERATIONS EVALUATION

The licensee did not propose any changes that affect the staff's conduct of operations evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required.

11 RADIATION PROTECTION EVALUATION

The licensee did not propose any changes that affect the staff's radiation protection evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required because the radiation source terms are determined to be bounded by the previous approved contents.

12 QUALITY ASSURANCE EVALUATION

The licensee did not propose any changes that affect the staff's quality assurance evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required.

13 DECOMMISSIONING EVALUATION

The licensee did not propose any changes that affect the staff's decommissioning evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required.

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

The licensee did not propose any changes that affect the staff's waste confinement and management evaluation provided in the SERs supporting the original license issued on March 22, 2004, Amendment No. 1, February 10, 2010, and Amendment No. 2, January 19, 2012. Therefore, the staff determined that a new evaluation was not required.

15 ACCIDENT ANALYSIS EVALUATION

The licensee performed accident analyses for the changes to DC ISFSI requested in LAR 12-003. These analyses are evaluated by the staff in SER Sections 5, 6, and 8.

15.1 FINDINGS

F.1 The analyses of off-normal and accident events and conditions show that the design changes provided by LAR 12-003 would allow the DC ISFSI to meet the requirements of 10 CFR 72.122, will not exceed the radiation exposure limits to radiation workers or the public as specified in 10 CFR Part 20, and will not endanger the public health and safety.

16 TECHNICAL SPECIFICATIONS

16.1 Review Objective

The objective of this review is to determine whether the changes to the operating controls and limits or the TS for the DC ISFSI would meet the requirements of 10 CFR Part 72 under the proposed amendment. The licensee's proposed technical and editorial TS changes are identified in Section 1, paragraph 2 of this SER. Specifically, the proposed changes were reviewed to ensure that they support the fuel loading changes requested by the applicant. The technical and safety aspects of these changes were evaluated by the staff in Sections 5, 6, and 8 of this SER and were found to be acceptable.

16.2 Findings

F16.1 The staff finds that the conditions for use at the DC ISFSI identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The proposed TS changes provide revised conditions that provide reasonable assurance that the DC ISFSI (as operated within the conditions) would continue to allow safe storage of spent fuel.

17 ENVIRONMENTAL CONSIDERATION

The staff performed an environmental evaluation of LAR 12-003 along with the categorical exclusion requirements of 10 CFR 51.22(c)(11), which provides for a categorical exclusion for:

> Issuance of amendments to licenses for fuel cycle plants and radioactive waste disposal sites and amendments to materials licenses identified in § 51.60(b)(1) which are administrative, organizational, or procedural in nature, or which result in a change in process operations or equipment, provided that (i) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, (ii) there is no significant increase in individual or cumulative occupational radiation exposure, (iii) there is no

significant construction impact, and (iv) there is no significant increase in the potential for or consequences from radiological accidents.

The staff finds that the changes requested in LAR 12-003 are changes in process equipment where (i) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, (ii) there is no significant increase in individual or cumulative occupational radiation exposure, (iii) there is no significant construction impact, and (iv) there is no significant increase in the potential for or consequences from radiological accidents. The staff finds that LAR 12-03 meets the requirements for a categorical exclusion per 10 CFR 51.22(c)(11), and therefore, pursuant to § 51.22(b), an environmental assessment or environmental impact statement need is not required.

18 CONCLUSION

Based on its review of LAR-12-003, as revised and supplemented, the staff has determined that there is reasonable assurance that: (i) the activities proposed by the amendment can be conducted without endangering the health and safety of the public and (ii) these activities would be conducted in compliance with the applicable regulations of 10 CFR Part 72. The staff has further determined that the issuance of the amendment would not be inimical to the common defense and security. Therefore, the staff concludes that the amendment should be approved.

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Issued with Amendment No. 3 to Materials License No. SNM-2511

February 11, 2014