SAFETY EVALUATION REPORT

DOCKET NO. 72-26
PACIFIC GAS & ELECTRIC COMPANY
MATERIALS LICENSE NO. SNM-2511
AMENDMENT NO. 2

1 SUMMARY

By letter dated January 31, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110400377), as supplemented June 8 (ML11173A228), July 28 (ML11216A208), September 15 (ML11262A270), and November 22, 2011 (ML11333A063), Pacific Gas and Electric Company (PG&E) submitted license amendment request (LAR) 11-001 to the United States Nuclear Regulatory Commission (NRC) to amend Materials License No. SNM-2511 for the Diablo Canyon (DC) site 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.

LAR 11-001 proposed the following revisions to the technical specifications (TS):

a. TS 1.1, "Definitions" - revise to reflect the addition of high burnup fuel (HBF), neutron source assemblies (NSAs), and instrument tube tie rods (ITTRs) to the allowable loading contents.

b. TS 2.0, "Approved Contents" - revise Tables 2.1-1 through 2.1-7, and 2.1-10 to reflect the addition of high burnup fuel (HBF), neutron source assemblies (NSAs), and instrument tube tie rods (ITTRs) to the allowable loading contents.[1]

c. TS 2.0, "Approved Contents" - Table 2.1-7, revise Fuel Assembly Cooling and Maximum Decay Heat (Uniform Fuel loading) for a MPC - 32 to limit the decay heat load to 750 W per assembly for a canister containing HBF.[1]

d. TS 2.0, "Approved Contents" - revise to add new TS 2.3 and associated Table 2.3-1 to provide alternative calculations for burnup limits for fuel assemblies in a MPC-32 to allow the storage of HBF.

e. TS 3.1.1, "Multi-Purpose Canister" - revise to eliminate the vacuum drying option which is not allowed for HBF and to add a reference temperature of 70°F for the MPC Helium backfill pressure range.

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1 Revised by the licensee in its November 22, 2011, supplement.

f. TS 3.1.2, "Spent Fuel Storage Cask (SFSC) Heat Removal System," - revise to allow the HI-STORM 100 Shortened Anchored (100SA) overpack to be considered operable with up to 50 percent vent blockage (although removal of any blockage is still required on discovery).

g. TS 3.1.4, "Supplemental Cooling System [SCS],"- add to provide the conditions and criteria for the SCS.

h. TS 4.1.2b, "Design Features Important to Criticality Control," - revise to change the B4C content in METAMIC to ≤ 33.0 wt%.

i. TS 5.1.3b, "MPC and SFSC Loading, Unloading, and Preparation Program," - revise to delete the requirement for maintaining the annulus full during drying and to restore the requirement for maintaining the annulus full during reflood (unloading).

LAR 11-001 also proposes to revise the licensing basis from that documented in the DC ISFSI Final Safety Analysis Report Update (FSARU) to:

j. Upgrade the thermal analysis methodology to a three dimensional (3D) Computational Fluid Dynamics (CFD) model,

k. Remove the assumption of 100% fuel failure coincident with 100% vent blockage,

l. Change of some allowed component temperatures (peak cladding, concrete, overpack metal, transfer cask lid neutron shielding) in the thermal evaluation,

m. Reduce the required torque criteria for the MPC lift cleats, and

n. Add a new accident for loss of SCS to the design criteria for the SCS.

The licensee initially requested an exemption from the requirement of 10 CFR 72.236(f) to design spent fuel storage casks to provide adequate heat removal capacity without active cooling systems. The staff informed the licensee that this was not required for the DC site specific ISFSI, and is only required by facilities licensed under 10 CFR 72 Subpart K (general licensees). PG&E subsequently withdrew this request in its July 28, 2011, supplement.

2 BACKGROUND

The Diablo Canyon 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. Where applicable, the DC ISFSI Final Safety Analysis Report (FSAR) utilizes site-specific information previously presented in the DCPP FSAR.

The two reactor units of DCPP share a common fuel-handling building and auxiliary building as well as components of auxiliary systems. Each unit has a dedicated fuel-handling system and spent nuclear fuel pool. Both units share a single 125 ton-[113,398 kg]-capacity crane for fuel handling activities. Each reactor core contains 193 fuel assemblies, and both units are currently operating on 18 to 21 month refueling cycles. Typically, 76 to 96 spent nuclear fuel assemblies are permanently discharged from each unit during a refueling.

The DC ISFSI is designed to hold up to 140 storage casks. Based on the current fuel strategy and the principal use of the multi-purpose canister (MPC) that contains a maximum of 32 pressurized water reactor fuel assemblies (MPC-32), the DC ISFSI is capable of storing all of the spent nuclear fuel generated by the two DCPP reactors during the terms of its current operating licenses. 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, PG&E is licensed to use other MPC designs from the HI-STORM 100 Cask System, including the MPC-24, MPC-24E, and MPC-24EF designs.

3 REVIEW CRITERIA

The staff evaluation of the proposed changes are 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 portions of the license, TS, and the FSAR 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.

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 (SER)s supporting the original license issue on March 22, 2004, and Amendment No. 1 issued on February 10, 2010. Therefore, the staff determined that a new evaluation was not required.

5 STRUCTURAL EVALUATION

The proposed TS changes do not impact the original structural evaluation. Therefore an evaluation was not required for the TS changes.

The licensee provided the following proposed structural licensing basis change for staff evaluation:

m. Reduce the required torque criteria for the MPC lift cleats.

The licensee requested to change the criteria in the FSAR from a specific torque value to “wrench tight.” The change was requested to reduce personnel dose during the MPC transfer evolution. The licensee stated that this change had been previously evaluated and found acceptable by the staff in the approval of HI-STORM 100 Cask System, Certificate of Compliance (CoC) No. 1014, Amendment No. 4, supported by the HI-STORM 100 FSAR, Rev. 5.

The staff determined that the DC MPC and lift cleats are the same equipment used in the HI-STORM 100 Cask System, CoC No. 1014. The lift cleat procedure and torque value was revised and approved by the staff in CoC 1014, Amendment No. 4. Therefore, the staff finds this change acceptable.

6 THERMAL EVALUATION

6.1 Review Objective

The objective of the thermal evaluation is to confirm that the decay heat removal system remains capable of reliable operation so that the temperatures of materials using SSCs ITS and fuel assembly cladding material remain within the allowable limits provided in NUREG -1567, and Interim Staff Guidance (ISG) -11 under normal, off-normal, and accident conditions.

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

- TS 3.1.1, "Multi-Purpose Canister (MPC)," - revise to eliminate the vacuum drying option, which is not allowed for High Burnup Fuel (HBF), and to add a reference temperature of 70°F for the MPC Helium backfill pressure range.

- TS 3.1.2, "Spent Fuel Storage Cask (SFSC) Heat Removal System," - revise to allow the HI-STORM 100SA overpack to be considered operable with up to 50 percent vent blockage (although removal of any blockage is still required on discovery).

- TS 3.1.4, "Supplemental Cooling System,"- add to provide the conditions and criteria for the SCS.

- TS 5.1.3b, "MPC and SFSC Loading, Unloading, and Preparation Program," - revise to delete the requirement for maintaining the annulus full during drying and to restore the requirement for maintaining the annulus full during reflood (unloading).

The staff thermal evaluation also reviewed the revised licensing basis in the provided DC ISFSI FSARU that proposed to:

- Upgrade the thermal analysis methodology to 3-D CFD model.

- Remove the assumption for 100 percent fuel failure coincident with 100 percent vent blockage.

- Change of some allowed component temperatures in the thermal evaluation (peak cladding, concrete, overpack metal, transfer cask lid neutron shielding) in the thermal evaluation.

- Add a new accident for loss of SCS to the design criteria for the SCS.

6.2 Evaluation

6.2.1 Licensee Thermal Models

The licensee performed a site specific thermal analysis to verify that the modified MPC-32 design is within the limits previously evaluated by the staff for CoC No. 1014, Amendment No. 3. This analysis demonstrated that for all conditions of system operation with a design basis heat load of 24 kW; the modified design remained within the limits previously evaluated. In LAR 11-001 the site specific DC ISFSI thermal analysis was updated to a 3-D CFD analysis, and the analysis was modified to address the storage of high burnup fuel consistent with the evaluation guidance in ISG -11, Rev. 3. This licensee's analysis in LAR 11-001 demonstrated that fuel cladding temperatures were maintained below NUREG -1567 acceptance limits for all operating conditions for the DC ISFSI. The HI-STORM 100 Cask System FSAR performed the evaluation using the SCS only for a helium-filled MPC containing HBF loaded in the HI-TRAC transfer canister. The SCS is used to maintain the temperature of the MPC shell at a temperature that ensures that the maximum temperature of the fuel cladding does not exceed its long term normal operating limits provided in ISG - 11. As part of its analysis, the licensee updated some of the individual component temperature limits were updated to those approved in CoC No. 1014, Amendment No. 5. 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 media using the flow impedance properties computed using a CFD approach. 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 used in the 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 100SA overpack are modeled explicitly. The model includes all 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 Fluent. 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 flow resistance of Westinghouse 17x17 fuel assemblies calculated using CFD methods are used in the thermal analyses. The fuel assembly region of the 3-D model is separated into three axial sections: bottom inactive length, active length, and top inactive length. The top inactive length in the DC MPC-32 is shorter than the design basis top inactive length. However, the DC 17X17 fuel assembly is modeled with an overstated bottom inactive length. The active fuel length is unchanged. As the bottom inactive length has the highest flow resistance of the three axial sections the licensee determined the total flow resistance used is conservative.

The CTF is a steel cylinder backed by concrete. In the licensee's model it is 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 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 solid zones that represent the inner shell, the radial lead shield, the outer shell, the water jacket and the enclosure shell.

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 which are described as Scenario 1 and Scenario 2 in the following table:

	Q (kW)	Storage pattern	Absolute operating pressure (atm)
Scenario 1	24	Uniform	4.8
Scenario 2	36.9	Regionalized	7

The licensee requested a total heat load limit of 24 kW for the MPC-32 loaded with HBF. The licensee stated that scenario 2 was evaluated in the application because it bounded Scenario 1 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 a steady state analysis that demonstrated that the peak cladding temperature remained below allowable limits provided in ISG -11 indefinitely.

<u>6.2.2.1 Thermal Evaluation for Normal Storage</u>

The licensee performed an evaluation for a maximum uniform heat load of 24 kW, a helium backfill of 29.3 psig at 70°F, and an MPC operating pressure of 4.8 atmospheres. The licensee stated that 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.

<u>6.2.2.2 Thermal Evaluation for the HI-STORM 100SA in the Cask Transfer Facility</u>

The licensee stated that this condition 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 DC ISFSI heat load described in sections 6.2.1 and 6.2.2. The licensee's analysis show the fuel cladding temperature and MPC and overpack temperatures remained below their respective long-term normal operating temperature limits provided in ISG -11. Also, the licensee's calculated maximum operating pressure remained below the normal design pressure limit in the DC ISFSI FSAR. The licensee determined 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 24 kW.

6.2.2.3 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 2 is the limiting fuel storage configuration during transfer operations. The licensee's results show that the peak fuel cladding temperature during normal on-site transfer conditions remained below its temperature limit for moderate burnup fuel, but exceeds the ISG -11 allowable limit for HBF. For HBF the licensee stated that it will use an SCS to maintain the maximum cladding temperature for HBF below the 400°C temperature limit guidance provided in ISG -11 for any MPC that contains one or more HBF assemblies. Also, the licensee stated that the results of its analysis indicate the maximum operating pressure remains below the normal design pressure limit in this configuration.

6.2.2.4 Thermal Evaluation During Off-Normal and Accident Conditions

The licensee considered three off-normal events for the DC ISFSI storage configuration: off-normal ambient temperature, off-normal pressure, and partial blockage of air inlets. The licensee's analysis shows that all the MPC and HI-STORM 100SA component temperatures remain below their temperature limits for the considered off-normal events. Also, the calculated MPC pressure under off-normal ambient temperature is below the off-normal design pressure for the considered off-normal events.

The licensee considered four accident events 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 the accident limit guidance of 570°C (1078°F) provided in NUREG -1567. 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 demonstrated that for a blockage duration of 32 hours, all MPC and overpack components temperatures remain below the accident limit guidance of 570°C (1078°F) provided in NUREG -1567. Also, the calculated pressure remains 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 the accident limit guidance of 570°C (1078°F) provided in NUREG -1567 and all the MPC and HI-TRAC overpack component temperatures remain also below their respective temperature limits as provided in NUREG -1567. The pressure analysis also shows that the maximum pressure remains below its accident limit. For the fire accident 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 calculation shows that fuel cladding and all component temperatures are below accident temperature limits provided in NUREG -1567. The calculation shows the calculated pressure remains well below the accident limit provided in the DC FSAR. 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.

6.3 Staff Confirmatory Analyses and Conclusions

The staff reviewed LAR 11-001 and determined that the updated FSAR sections include accurate information to enable the staff to evaluate the thermal effectiveness of SSCs important to safety and make a determination on the acceptability of the proposed TS changes and licensing bases. The staff evaluation of the licensee's thermal models discussed above 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, based on its review of the design details, that the licensee used proper material properties and boundary conditions. The staff determined that the licensee's selected code models accurately reflect the specific design parameters. In addition, the staff determined that the licensee's assumptions and modeling parameters are consistent with guidance in NUREG-1567, and therefore determined the assumptions are adequate for the flow and heart transfer characteristics prevailing in the HI-STORM 100SA geometry and analyzed conditions. The staff reviewed the licensee's engineering drawings to verify that adequate geometry dimensions were translated to the analysis models. The staff also reviewed material properties presented in the FSAR to verify that they were appropriately referenced and used. Based on its review of the licensee's thermal models, the staff determined that the licensee's utilization of computer modeling is consistent with the guidance in NUREG-1567 and ISG - 21, "Use of Computational Software," and is therefore acceptable.

The staff performed an independent confirmatory analysis. The staff compared the results of its confirmatory analysis to the licensee's sensitivity analysis, and determined that the licensee's analysis was 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

Therefore, based on its review and the calculations described above, the staff finds the DC ISFSI thermal analysis and conclusions acceptable and that the DC ISFSI will continue to safely store spent nuclear fuel within TS parameters.

6.4 Evaluation Findings

F6.1 The submitted revised FSAR Chapter 2 describes SSCs important to safety to enable an evaluation of their thermal effectiveness. SSCs important to safety remain within their operating temperature ranges as provided in NUREG-1567 and ISG -11.

F6.2 The DC ISFSI continues to maintain heat-removal capability having verifiability and reliability consistent with its importance to safety as required by 10 CFR 72.128. Except during short-term operations, the cask is designed to provide adequate heat removal capacity without active cooling systems. The short term operations do not cause the fuel to exceed its design limits.

F6.3 The spent fuel cladding is protected against degradation leading to gross ruptures under long-term storage by maintaining cladding temperatures below 752°F (400°C) in a He environment. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal as required by 10 CFR 72.122(h)(1).

F6.4 The spent fuel cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining cladding temperatures below 1058°F

(570°C) in a He environment as required by 10 CFR 72.122(h)(1). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F6.5 The thermal design of the DC ISFSI complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, NUREG 1567, ISG -11, and ISG -21.

7 SHIELDING AND RADIATION PROTECTION EVALUATION

7.1 Background

This proposed license amendment would expand the allowable contents and loading capabilities for the DC ISFSI. Specifically, it would: allow loading of high burnup fuel (up to 68,200 MWD/MTU), ITTRs, and NSAs. The staff evaluated and approved ITTRs and NSAs as allowable contents to be loaded in approved MPCs under CoC No. 1014 Amendment Nos. 6 and 3, respectively. These changes modify the licenses original shielding and radiation protection analyses.

The licensee's shielding and radiation protection analysis includes a calculation of the dose rates on the ISFSI containing a full complement of HI-STORM 100SA overpacks, the CTF, and the 125-ton HI-TRAC transfer cask. Occupational exposures during loading and unloading operations of a HI-STORM I00SA overpack and maintenance and surveillance operations around the ISFSI are estimated in the licensee's submittal.

7.2 Review Objective

The objective of the staff's shielding and radiation protection evaluation is to determine that the shielding design features of the proposed DC ISFSI changes continue to meet 10 CFR Part 20 (10 CFR 20.1201, 10 CFR 20.1301, 10 CFR 20.1502) and 10 CFR Part 72 (10 CFR 72.104(a), 10 CFR 72.106(b)) limits for radiation protection to workers and to the public. These criteria establish the limits for dose due to direct radiation and radioactive effluents resulting from all ISFSI operations. The radiation protection evaluation includes a review of the information in Chapter 7, "Radiation Protection," of the DC ISFSI FSAR that may be affected by LAR 11-001.

The following proposed revisions to the TS are applicable to the shielding and radiation protection evaluation.

b. TS 2.0, "Approved Contents" - revise Tables 2.1-1 through 2.1-7, and 2.1-10 to reflect HBF selection criteria and the addition of NSAs and ITTRs.

d. TS 2.0, "Approved Contents" - revise to add new TS 2.3 and associated Table 2.3-1 to provide alternative calculations for burnup limits for fuel assemblies in a MPC-32 to allow the storage of HBF.

7.3 Staff Confirmatory Calculations

The staff performed confirmatory annual dose calculations for an individual located on the nearest boundary of the ISFSI controlled area and for the resident nearest to the ISFSI provided in the licensee supplied Holtec Report No: HI-2002563. Based on staff confirmatory calculations the staff finds the on-site dose rates and off-site dose rates provided by the licensee within 10 CFR Part 20 and 10 CFR Part 72 limits and are therefore acceptable.

7.4 Evaluation and Conclusions

The gamma and neutron source terms are provided in Section 7.2 of the DC ISFSI FSAR. The licensee's radiation protection analyses for dose rates from direct radiation assumed that the HI-STORM 100SA overpacks contained MPC-32s completely loaded with fuel assemblies having identical burnup and cooling times. In the original analysis, the licensee assumed that the fuel burnup was 32,500 MWD/MTU with an initial cooling time of five years. To allow the DC ISFSI to be loaded with HBF, the licensee performed a shielding analysis. The burnup was increased to 69,000 MWD/MTU for assemblies of 4.8 wt% U-235 initial enrichment, with an initial cooling time of five years. The licensee demonstrated that the dose rates calculated for an overpack for the original shielding analyses for the MPC-32 bound the dose rates calculated for a HI-STORM 100SA overpack containing an MPC-24, MPC-24E, or MPC-24EF.

The licensee stated that the allowed loading of fuel was based on CoC No. 1014, Amendment No. 3, and the revised dose analysis was performed using CoC No. 1014, Amendment No. 5, source terms supported by the HI-STORM 100 FSAR, Rev. 7. The licensee stated that this resulted in higher calculated personnel doses since the CoC No. 1014, Amendment No. 5, fuel was allowed to be loaded at a higher heat load which reduced the time allowed for radioactive decay of the fission products in the fuel. The licensee calculated the source terms for the analyses using the same methods described in the HI-STORM 100 Cask System FSAR, Rev. No. 7. In the revised licensee analysis for HBF, the transfer cask shielding analysis used the MPC-24 analysis from the HI-STORM 100 FSAR Rev. 7 that provided a burnup of 75,000 MWD/MTU and a cooling time of 5 years. To estimate the dose for an MPC-32, the licensee multiplied by the ratio of assemblies contained, which provided conservative results since it did not take into consideration the increased self-shielding in the MPC-32. It was demonstrated in the original shielding analysis that the dose rates on the surface of the transfer cask used this burnup and cooling time and bound the dose rates using other allowable burnup and cooling times. It was also demonstrated that the dose rates from a transfer cask containing an MPC-24 bounded the dose rates from a transfer cask containing an MPC-32. The burnup and cooling time combination was selected from the HI-STORM 100 FSAR, Rev. 7. The licensee determined that the site boundary dose rates for high burnup fuel in the HI-STORM 100SA overpack containing MPC-32 at the DC ISFSI met the limits in 10 CFR 72.104 for 2080 hr/year.

The staff evaluated the licensee's analyses of the bounding radiation source terms for the DC ISFSI. The staff determined that the description of radiation sources and calculation methods of the shielding analysis are consistent with the information provided in HI-STORM 100 System FSAR, Rev. 7 which used a burnup of 75,000 MWD/MTU and cooling time of five years, and therefore are acceptable.

DC calculated the maximum allowable fuel assembly average burnup for a given minimum enrichment as described below for minimum cooling times between 5 and 20 years using the maximum permissible decay heat used by DC. Different fuel assembly average burnup limits may be calculated for different minimum enrichments for use in choosing the fuel assemblies to be loaded into a given MPC.

DC used the following equation to calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 5 and 20 years using the following equation below:

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E\,235)^2] + (E \times q \times E235) + (F \times q^2 \times E235) + G$$

Where:

Bu = Maximum allowable average burnup per fuel assembly (MWD/MTU)
q = Maximum allowable decay heat per storage location, in kilowatts,
E235 = Minimum fuel assembly average enrichment (wt% 235U)
A through G = Coefficients from CoC No. 1014, Amendment No. 3.

NSAs have been added to the allowable contents at the DC ISFSI. During in-core operations, the stainless steel and Inconel portions of the NSAs become activated, producing a significant amount of Co-60. Using the masses of steel and Inconel for the NSAs, the staff determined that the total activation of a primary or secondary source was bounded by the total activation of a BPRA as described in HI-STORM 100 FSAR.

The staff reviewed the calculations described above, and found that DC assessed the dose rates in accordance with NUREG-1567 guidance. The staff also reviewed the licensee's shielding details and found that the description met the 10 CFR 72.126(a)(6) requirement to provide a radiation protection system for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials and to design, fabricate, locate, shield, control and test structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure to control external and internal radiation exposures. Based on a review of the licensee's design drawings, analysis models, and NUREG-1567, the staff determined that the radiation protection systems were appropriately modeled in the shielding analysis. The staff therefore finds that the DC ISFSI will continue to meet 10 CFR Part 20 and 10 CFR Part 72 limits for radiation protection to workers and to the public within the conditions of the TS.

7.5 Evaluation Findings

The staff made the following findings regarding the radiation and shielding evaluation of the DC ISFSI:

F7.1 The design of the shielding system of the DC ISFSI continues to satisfy the 10 CFR 72.126(a)(6) requirement to shield personnel from radiation exposure.

F7.2 The design of the DC ISFSI continues to provide acceptable means for controlling occupation radiation exposures within the limits of 10 CFR 20.1201 and meeting the objective of maintaining exposures as low as reasonably achievable in compliance with 10 CFR 72.24(e).

F7.3 The design of the DC ISFSI continues to provide acceptable means for controlling exposures of the public to direct and scattered radiation within the limits given in 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.4 The design of the DC ISFSI continues to provide suitable shielding for radioactive protection during normal and accident conditions in compliance with 10 CFR 72.128(a)(2).

8 CRITICALITY EVALUATION

8.1 Background

Four of the changes affecting the criticality safety of the DC ISFSI proposed in LAR 11-001 are intended to conform with changes made to CoC No. 1014 and subsequent amendments that have previously been evaluated and approved by the NRC. This includes the addition of NSAs and ITTRs as allowable contents that were approved under CoC No. 1014, Amendments No. 3 and 6, respectively.

One change proposed by the licensee in this amendment that could potentially affect the criticality safety and was not previously evaluated by the staff is the inconsistency discovered by the licensee between the current DC ISFSI license and CoC No. 1014 pertaining to the B_4C content in METAMIC. The METAMIC neutron-absorbing panels incorporated into the fuel basket maintain criticality safety of the HI-STORM 100 Cask System. TS 4.1.2b currently has a limit on the B_4C content of "< 33.0 wt%". The licensee proposed revising this limit to "≤ 33.0 wt%"to provide consistency with the CoC 1014 TS.

8.2 Review Objective

The objective of the criticality evaluation is to ensure that the stored materials remain subcritical under normal, off-normal, and accident conditions during all operations, transfers, and storage at the DC ISFSI.

The following proposed technical revisions to the TS are applicable to the criticality evaluation.

a. TS 1.1, "Definitions" - revise to reflect terms to modify high burnup fuel (HBF) selection criteria and the addition of neutron source assemblies (NSAs), and instrument tube tie rods (ITTRs).

b. TS 2.0, "Approved Contents" - revise Tables 2.1-1 through 2.1-7, and 2.1-10 to reflect HBF selection criteria and the addition of NSAs and ITTRs.

c. TS 2.0, "Approved Contents" - Table 2.1-7, revise Fuel Assembly Cooling and Maximum Decay Heat (Uniform Fuel loading) for a MPC - 32 to limit the decay heat load to 750 W per assembly for a canister containing HBF.

d. TS 2.0, "Approved Contents" - revise to add new TS 2.3 and associated Table 2.3-1 to provide alternative calculations for burnup limits for fuel assemblies in a MPC-32 to allow the storage of HBF.

The following proposed editorial change is applicable to the criticality evaluation.

h. TS 4.1.2b, "Design Features Important to Criticality Control," - revise to change the B4C content in METAMIC to ≤ 33.0 wt%.

8.3 Evaluation and Conclusions

The staff evaluated proposed technical changes a., b., c., and d. and determined that they are applicable to the DC ISFSI and consistent with previous NRC approved changes in CoC No. 1014, Amendment Nos. 3 and 6. The staff therefore finds the requested changes acceptable. With regard to the one change that was not previously reviewed by the staff, proposed change “h”, the staff determined that this change is editorial in nature and intended to correct an inconsistency between the current DC ISFSI license and CoC No. 1014.

Because this change is editorial in nature and does not affect criticality safety, the staff finds this change acceptable.

8.4 FINDINGS

F8.1 The proposed designs and proposed use of the DC ISFSI handling, packaging, transfer, and storage systems for the radioactive materials to be stored ensure that the materials will remain subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The margins of safety are acceptable because they are within the guidance provided in NUREG-1567, and comply with 10 CFR 72.124(a) and 10 CFR 72.124(b).

9 CONFINEMENT EVALUATION

9.1 Review Objective

The objective of the confinement evaluation is to determine that the DC ISFSI changes proposed in LAR 11-001 continue to ensure radiological releases to the environment will be within acceptable limits, and that the spent fuel cladding will be sufficiently protected during storage operations against degradation.

The following proposed technical revisions to the TS are applicable to the confinement evaluation.

a. TS 1.1, "Definitions" - revise to include terms to reflect high burnup fuel (HBF) selection criteria and the addition of neutron source assemblies (NSAs), and instrument tube tie rods (ITTRs).

b. TS 2.0, "Approved Contents" - revise Tables 2.1-1 through 2.1-7, and 2.1-10 to reflect HBF selection criteria and the addition of NSAs and ITTRs.

c. TS 2.0, "Approved Contents" - Table 2.1-7, revise Fuel Assembly Cooling and Maximum Decay Heat (Uniform Fuel loading) for a MPC - 32 to limit the decay heat load to 750 W per assembly for a canister containing HBF.

d. TS 2.0, "Approved Content" - revise to add new TS 2.3 and associated Table 2.3-1 to provide alternative calculations for burnup limits for fuel assemblies in a MPC-32 to allow the storage of HBF.

9.2 Evaluation and Conclusions

The staff evaluated the DC ISFSI LAR 11-001 changes to assess their impact on the confinement design and performance. The staff compared the proposed changes to the design previously evaluated for DC ISFSI License, Amendment No. 1. In its evaluation of Amendment No. 1, the staff determined that the DC ISFSI would ensure radiological releases to the environment are within acceptable limits, and that the spent fuel cladding would be sufficiently protected during storage operations against failure and degradation. The staff determined that the changes requested in LAR-11-001 do not affect the staff's previous confinement evaluation. Therefore the staff finds these changes acceptable.

9.3 Evaluation Findings

F9.1 The design and proposed operations of the DC ISFSI continue to provide adequate measures for protecting the spent fuel cladding against degradation that might otherwise lead to gross ruptures of the material to be stored, in compliance with 10 CFR 72.122(h)(1).

F9.2 The MPC remains in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria continue to remain acceptable. This finding is reached on the basis of a review that considered the regulation itself, NUREG-1567, and ISG - 18, Rev. 1.

10 CONDUCT OF OPERATIONS EVALUATION

The licensee provided the following proposed conduct of operations licensing basis change for staff evaluation:

m. Reduce the required torque criteria for the MPC lift cleats.

The licensee requested to change the criteria in the FSAR from a specific torque value to "wrench tight." The change was requested to reduce personnel dose during the MPC transfer evolution. The licensee stated that this change had been previously evaluated and found acceptable by the staff in the approval of HI-STORM 100 Cask System, Certificate of Compliance (CoC) No. 1014, Amendment No. 4, supported by the HI-STORM 100 FSAR, Rev. 5.

The staff determined that the DC MPC and lift cleats are the same equipment used in the HI-STORM 100 Cask System, CoC No. 1014. The lift cleat procedure and torque value was revised and approved by the staff in CoC 1014, Amendment No. 4. Therefore, the staff finds this change acceptable.

RADIATION PROTECTION EVALUATION

Refer to SER Section 7.

11 QUALITY ASSURANCE EVALUATION

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

12 DECOMMISSIONING EVALUATION

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

13 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 issue on March 22, 2004, and Amendment No. 1 issued on February 10, 2010. Therefore, the staff determined that a new evaluation was not required.

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14 ACCIDENT EVALUATION

The licensee performed accident analyses for the changes to DC ISFSI requested in LAR 11-001. These analyses were evaluated by the staff in SER Sections 6, 7, 8, and 9.

15.1 FINDINGS

F.1 The analyses of off-normal and accident events and conditions and reasonable combination of these and normal conditions show that the design changes provided by LAR 11-001 will continue to allow the DC ISFSI to meet the requirements of 10 CFR 72.122 and will not endanger the public health and safety.

15 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 continue to meet the requirements of 10 CFR Part 72. The applicant proposed technical and editorial TS changes a. through i. that are identified in Section 1, paragraph 2 of this SER. Specifically, the proposed changes were reviewed to ensure that they support the equipment changes requested by the applicant. The technical and safety aspects of these changes were evaluated by the staff in sections 6, 7, 8, and 9 of this SER and were found to be acceptable.

16.2 Findings

F16.1 The staff concludes 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 reasonable assurance that the DC ISFSI will continue to allow safe storage of spent fuel. This finding is based on consideration of the regulation, NUREG-1567, and a review of staff approved TS changes in CoC 1014, Amendments No. 3 and 6.

17 ENVIRONMENTAL CONSIDERATION

In its January 31, 2011, application, the licensee provided the following to address 10 CFR 51.41 considerations:

> "The proposed changes do not significantly change the type or significantly increase in the amounts of any effluents that may be released offsite. In addition, there is no significant increase in individual or cumulative occupational radiation exposure. The proposed changes do not involve construction of any kind. Therefore, there is no significant construction impact. The proposed changes do not involve an increase in

> the potential for or consequences from radiological accidents. There is an increase in the total offsite dose from normal operations as a result of the use of the new dose analysis supporting loading of HBF. However, the total offsite doses remains below the 10 CFR 72.104 limits and are considered acceptable."

LAR-11-001 requests changes to the MPC lift cleat torque value and loading of HBF, ITTRA, and NSAs. In addition, LAR-11-001 requests changes to conform to CoC 1014. Therefore, staff has determined that the changes requested result in a change in process operations and equipment. The staff evaluations and findings provided in SER Sections 7 and 9 have determined that a) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, b) there is no significant increase in individual or cumulative occupational radiation exposure, and c) there is no significant increase in the potential for or consequences from radiological accidents. Staff evaluations in SER Sections 5, 6, and 8 have determined that there is no significant increase in the potential for or consequences from radiological accidents. Additionally the staff has determined that there are no construction activities associated with LAR 11-001. Therefore the staff finds that LAR 11-001 meets the requirements of 10 CFR 51.22(c)(11) for a categorical exclusion from the requirement to prepare an environmental assessment or environmental impact statement.

18 CONCLUSION

Based on its review of LAR-11-001, as revised and supplemented, the staff has determined that there is reasonable assurance that: (i) the activities authorized by the amended license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of 10 CFR Part 72. The staff has further determined that the issuance of the amendment will not be inimical to the common defense and security. Therefore, the amendment should be approved.

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Issued with Materials License No. SNM-2511

January 19, 2012