SAFETY EVALUATION REP

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

1 SUMMARY

By letter dated January 31, 2011, as supplemented June 8, July 28, September 15, and November 22, 2011, 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 include terms in support of 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 support HBF selection criteria and the addition of NSAs and ITTRs.[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," - delete reference to Table 2.1-9 as regionalized loading of high burn-up fuel is not being requested and changed the watts in the sample to 750 to be consistent with LAR 11-001.[1]

e. 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.

f. TS 3.1.1, "Multi-Purpose Canister (MPC)," - 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.

1 Revised by the licensee in its November 2P2, 2011, supplement.

g. 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).

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

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

j. TS 5.1.3b, "MPC and SFSC Loading, Unloading, and Preparation Program," - revise to delete the requirement for maintaining the annulus full during vacuum 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:

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

l. remove the assumption of 100% fuel failure coincident with 100% vent blockage,

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

n. reduce the required torque criteria for the MPC lift cleats, and

o. add a new accident for loss of SCS to the design criteria for the SCS.

The licensee initially requested an exemption from the requirements of 10 CFR 72.236(f) to allow use of a nonpassive SCS. The staff determined that this was not required for the DC site specific ISFSI. 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 Diablo Canyon ISFSI provides additional spent nuclear fuel storage capacity to DCPP beyond 2006, when the wet pool storage was near full capacity. Where applicable, the Diablo Canyon 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 Diablo Canyon 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 Diablo Canyon ISFSI is capable of storing all of the spent nuclear fuel generated by the two DCPP reactors during the terms of their current operating licenses. The Diablo Canyon ISFSI consists of the Holtec HI-STORM 100 cask storage system, a Cask Transfer Facility (CTF), an onsite cask transporter, and the storage pads. In addition, to accommodate spent nuclear fuel generated during the ISFSI licensed period, as well as any damaged fuel assemblies, debris, and nonfuel hardware, PG&E may use three 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's 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 utilized the guidelines provided in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities” in conducting its evaluation. The staff’s evaluation focused only on changes to SNM-2511 and associated TS requested in the application and 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 as described below for each of the proposed changes.

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

The proposed changes do not impact the original SSC and design criteria evaluation. Therefore an evaluation was not required.

5 STRUCTURAL EVALUATION

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

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 used SSCs important to safety 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.

- 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,"- added 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 vacuum drying and to restore the requirement for maintaining the annulus full during reflood (unloading).

The thermal evaluation also reviewed the revised licensing basis as documented in the provided DC ISFSI FSARU which proposed to:

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

- Remove the requirement 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.)

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

6.2 Evaluation

Thermal Models

The licensee performed a site specific thermal evaluation to verify that the modified MPC-32 design was in compliance with 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 required temperature limits were met. In support of 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 per the evaluation guidance of Interim Staff Guidance (ISG) -11, Rev. 3. This analysis demonstrated that fuel cladding temperatures met the requirements for all conditions, although a supplemental cooling system (SCS) was required for a helium-filled MPC containing HBF loaded in the HI-TRAC. 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. As part of this new analysis 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 explicitly. 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 Diablo Canyon 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 total flow resistance used is conservative.

The cask transfer facility (CTF) is a steel cylinder backed by concrete. 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. This maximizes the computed temperatures of both the HI-STORM 100SA and the CTF, and is a conservative assumption.

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.

The staff evaluated the description of the thermal models used to perform the thermal evaluation of the DC ISFSI system in the storage configuration, CTF configuration, and transfer configuration and found them acceptable and adequate to represent the key heat transfer and fluid flow phenomena of the system.

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. 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 by a steady state analysis that demonstrated that the peak cladding temperature remained below allowable limits indefinitely.

Thermal Evaluation for Normal Storage

This evaluation was performed to support the DC ISFSI license 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 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.

Thermal Evaluation for the HI-STORM 100SA in the Cask Transfer Facility

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 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 provided in DC ISFSI FSAR. Therefore, 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.

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 allowable limit for high-burnup fuel. For HBF the licensee will use a supplemental cooling system (SCS) to maintain the maximum cladding temperature for HBF below the 400°C temperature limit for any MPC that contains one or more HBF assemblies. Also, the results of the analysis indicate the maximum operating pressure remains below the normal design pressure limit in this configuration.

Based on the evaluation of LAR 11-001, the staff finds the description, assumptions, and analysis results of normal storage, transfer operations, and CTF configuration acceptable.

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 show 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 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 demonstrated that for a blockage duration of 32 hours, all MPC and overpack components temperatures remain below temperature limits. 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 all predicted temperatures and pressure remain below their accident limit.

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 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 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.

Based on the review and evaluation of the application, the staff finds the description, assumptions, and analysis results of off-normal and accident events during storage and transfer operations acceptable.

6.3 Confirmatory Analysis

The staff evaluated the licensee's thermal models used in the analyses. The staff checked the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The staff verified that the licensee's selected code models and assumptions were adequate for the flow and heart transfer characteristics prevailing in the HI-STORM 100SA geometry and analyzed conditions. The engineering drawings were also consulted 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 ensured that the licensee performed appropriate sensitivity analysis calculations to obtain mesh-independent results that provided bounding predictions for all analyzed conditions during normal storage, transfer operations, and off-normal and accident events. Finally, the staff ensured that the licensee provided proposed updated FSAR sections that included complete and accurate information that allowed the staff to make a safety determination on the acceptability of the DC ISFSI.

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.

F6.2 The DC ISFSI continues to maintain heat-removal capability having verifiability and reliability consistent with its importance to safety. 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). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

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). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F6.5 The staff finds that the thermal design of the DC ISFSI is in compliance 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, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7 SHIELDING AND RADIATION PROTECTION EVALUATION

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), fuel assemblies susceptible to top nozzle cracking that have been structurally repaired using instrument tube tie rods (ITTRs), and neutron source assemblies (NSAs). The staff evaluated and approved ITTRs and NSAs as allowable contents for HI-STORM 100 General Licensees under CoC No. 1014 Amendment Nos. 6 and 3, respectively. The changes affecting the shielding of the DC ISFSI proposed in this amendment relate to changes that have been made to the CoC No. 1014 and subsequent amendments that have already been reviewed and approved by staff.

7.1 Review Objective

The objective of the staff's radiation protection evaluation is to evaluate that the shielding design features of the proposed DC ISFSI changes continue to meet 10 CFR Part 20 and 10 CFR Part 72 limits for radiation protection to workers and to the public. These criteria establish the limits for direct radiation 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 radiation protection analysis includes calculation of the dose rates from the HI-STORM I00SA overpack on the ISFSI and CTF, the 125-ton HI-TRAC transfer cask, and the ISFSI site containing a full complement of HI-STORM 100SA overpacks. 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. The radiation protection review considered how the information in the application addressed the following regulatory requirements contained in 10 CFR Part 72:

- Paragraph 72.104(a) requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area be limited to 0.25 mSv (25 mrem) to the whole body, and 0.75 mSv (75 mrem) to the thyroid or 0.25 mSv (25 mrem) to any other critical organ.

- Paragraph 72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel, waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

For onsite dose rates, the following dose rate limits are used which are consistent with the requirements specified in I0 CFR Part 20:

- 5 rem/year for personnel with dose rate monitors (10 CFR 20.1201),
- 0.5 rem/year for personnel without dose rate monitors (10 CFR 20.1201 and 10 CFR 20.1502),
- 2 mrem in any one hour and 100 mrem/year for individual members of the public (10 CFR 20.1301)

The gamma and neutron source terms are provided in Section 7.2 of the Diablo Canyon 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, fuel burnup was assumed to be 32,500 MWD/MTU with an initial cooling time of 5 years. To allow the DC ISFSI to be loaded with HBF, the shielding analysis was reperformed in support of LAR 11-001. 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 5 years. It was demonstrated by the licensee that the dose rates calculated for an overpack containing the original shielding analyses for MPC-32 bound the dose rates calculated for a HI-STORM 100SA overpack containing an MPC-24, MPC-24E, or MPC-24EF.

The staff evaluated the licensee's analyses of the bounding radiation source terms for the DC ISFSI. The staff found the description of radiation sources and calculation methods of the shielding analysis to be consistent with the information provided in HI-STORM 100 System FSAR, Revision 7 which used a burnup of 75,000 MWD/MTU and cooling time of 5 years. The HI-STORM 100 System was approved by NRC for use under the general license provisions of 10 CFR Part 72. The staff found that combinations of enrichment, burnup, and cooling time for DC ISFSI spent nuclear fuel are conservative and was determined correctly. The shielding details are described in Section 7.3 of the FSAR. The staff evaluated the shielding details and found that the description satisfied the requirements of 10 CFR Part 72.126(a)(6) and provided reasonable assurance that the radiation protection systems were adequately modeled in the shielding analysis.

The estimates of dose rates and annual doses caused by direct neutron and gamma radiation at various on-site and off-site locations were provided in Sections 7.3.2.1, 7.3.2.2, 7.5.1, 7.5.3, and 7.5.4 of the FSAR.

The annual doses were calculated for an individual located on the nearest boundary of the ISFSI controlled area and for the resident nearest to the ISFSI's in Holtec Report No: HI-2002563. These calculations confirmed the on-site dose rates calculated by the licensee and also confirmed that the off-site dose rates would be less than the 0.25-mSv/yr [25-mrem/yr] whole-body dose allowable to a member of the public, as required by 10 CFR Part 72.104. Based on these confirmatory calculations, the staff finds that the licensee's shielding analysis is acceptable.

7.2 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 requirements of 10 CFR Part 72.126(a)(6).

F7.2 The design of the DC ISFSI continues to provide acceptable means for controlling occupation radiation exposures within the limits of 10 CFR Part 20.1201 and meeting the objective of maintaining exposures as low as reasonably achievable in compliance with 10 CFR Part 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 Part 72.104(a) and 10 CFR Part 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 Part 72.128(a)(2).

8 CRITICALITY EVALUATION

The majority of changes affecting the criticality safety of the DC ISFSI proposed in LAR 11-001 are consistent to those made to CoC No. 1014 and subsequent amendments that have already been evaluated and approved by staff. This includes the addition of NSAs and ITTRs as allowable contents, since they were approved under CoC No. 1014, Amendments No. 3 and 6, respectively.

The one change proposed by the licensee in this amendment that could potentially affect the criticality safety was 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 are required to maintain the 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. The staff agrees with the licensee that this change is editorial and represents no substantive change to the TS requirement and finds it acceptable.

8.1 FINDINGS

F8.1 The staff concludes that the criticality changes proposed in this amendment comply with the requirements of 10 CFR Part 72, and that the criticality design will continue to provide reasonable assurance that the cask storage system remains safe. This finding is reached based upon a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, appropriate risk-informed considerations, and accepted engineering practices.

9 CONFINEMENT EVALUATION

9.1 Review Objective

The objective of the confinement evaluation is to confirm the confinement features proposed in DC ISFSI 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 provision of LAR 11-001 was evaluated for its impact on confinement.

"Remove the requirement to determine a dose component from confinement boundary leakage if the boundary is tested to the leaktight criteria of ANSI N14.5-1997."

9.2 Evaluation

The licensee considered the confinement boundary of the MPCs that are loaded with HBF to be leak light. The factory shell welds and the vent and drain port cover plate welds will be tested to ANSI N14.5-1997 leak tight criteria. The lid-to-shell weld is a large, multi-pass weld which is placed and inspected consistent with the evaluation guidance of ISG-15. Therefore, consistent with the evaluation guidance of ISG-18, the staff finds that leakage from the lid and the vent and drain port cover is non-credible.

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 DC ISFSI confinement system will be tested to the leak tight criteria of ANSI N14.5-1997 during fabrication which demonstrates that it will maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F9.3 The staff finds that the design of the confinement system (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, appropriate regulatory guidance documents, applicable codes and standards, and accepted engineering practices.

10 CONDUCT OF OPERATIONS EVALUATION

The proposed changes do not impact the original conduct of operations evaluation. Therefore an evaluation was not required.

11 RADIATION PROTECTION EVALUATION

Refer to Section 7.

12 QUALITY ASSURANCE EVALUATION

The proposed changes do not impact the original quality assurance evaluation. Therefore an evaluation was not required.

13 DECOMMISSIONING EVALUATION

The proposed changes do not impact the original decommissioning evaluation. Therefore an evaluation was not required.

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

The proposed changes do not impact the original shielding evaluation. Therefore an evaluation was not required.

15 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 above.

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 acceptably meet the requirements of 10 CFR 72.122 and will not endanger the public health and safety.

16 TECHNICAL SPECIFICATIONS

16.1 Review Objective

The objectives of this review were to ensure that 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 evaluation was based on information provided by the applicant in LAR 11-001, a review of the supplied documentation and calculations, as well as consideration of accepted practices. Specifically, the proposed changes were reviewed to ensure that they acceptably supported the equipment changes requested by the applicant. The technical and safety aspects of these changes were evaluated by the staff in previous sections of this SER and were found to be acceptable. The applicant proposed technical and editorial TS changes, and these are identified in Section 1 of this SER.

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 the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

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