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SCE Selects Robust Underground System to Store San Onofre Used Nuclear Fuel

ROSEMEAD, Calif., Dec. 11, 2014 — Southern California Edison (SCE) has selected [Holtec International](#) to expand the San Onofre nuclear plant's storage of used nuclear fuel in a robust underground facility.

The contract with Holtec represents a major step in the decommissioning of the nuclear plant. It sets the stage to transfer San Onofre's used fuel from steel-lined concrete storage pools to steel-and-concrete-encased canisters, with a goal of completing the work by mid-2019.

"After reviewing leading designs with the [San Onofre Community Engagement Panel](#), we concluded this underground design is best suited to safely and securely store used nuclear fuel at San Onofre until the federal government removes the fuel from site, as required," said Chris Thompson, SCE vice president of Decommissioning. "Our decision to move expeditiously to transfer the fuel also reflects feedback from community leaders who prefer dry storage of used nuclear fuel."

Thompson noted the robust Holtec design exceeds California earthquake requirements and protects against hazards such as water, fire or tsunamis.

"I especially want to thank the Community Engagement Panel for its thoughtful questions and enormous time commitment during SCE's evaluation," said Thompson, noting that SCE ultimately focused on cask designs licensed by the [Nuclear Regulatory Commission](#) for both storage *and* transport of used nuclear fuel.

While dry storage of nuclear fuel is a proven technology used for almost three decades in the United States, Thompson said SCE will go beyond industry practices by partnering with the [Electric Power Research Institute](#) to develop new inspection techniques to monitor cask integrity.

Holtec's HI-STORM UMAX underground storage system features corrosion-resistant, stainless-steel fuel canisters topped with a 24,000-pound steel and concrete lid. The canisters will be encased in a concrete monolith. Holtec is a global supplier and has nuclear fuel storage systems at two other California locations, Humboldt Bay and Diablo Canyon. More information is available in this [fact sheet](#).

Thompson said engineering work begins immediately, followed by fabrication of canisters. Completion of the dry storage project facilitates major dismantlement work SCE plans to complete within 20 years.

SCE announced in June 2013 that it would [retire San Onofre Units 2 and 3](#), and begin preparations to decommission the facility. SCE has established core principles of safety, stewardship and engagement to guide decommissioning. For more information about SCE, visit www.songscommunity.com.

About Southern California Edison

An Edison International (NYSE:EIX) company, Southern California Edison is one of the nation's largest electric utilities, serving a population of nearly 14 million via 4.9 million customer accounts in a 50,000-square-mile service area within Central, Coastal and Southern California.

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EXHIBIT 6

San Onofre Nuclear Waste Problems

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Rear Admiral Len Hering Sr. USN (ret)

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INTRODUCTION

In August 2018, a near-accident during the loading of nuclear waste into dry storage triggered a federal investigation and brought new urgency to the debate of how best to store some of the most dangerous waste known to humankind – spent nuclear fuel. The San Onofre Nuclear Generating Station (S.O.N.G.S.) closed in 2012 after a number of serious failures. Since then, Southern California Edison and its contractor, Holtec International, built a concrete storage vault to hold 3.6 million pounds of nuclear waste in dry storage. That vault is footsteps from the rising Pacific Ocean. In our brief report, we explore the fatal flaws of this location and recommend moving the storage facility to a technically defensible storage facility at a significantly higher elevation with distance from the ocean. We address the inadequacy of the equipment used to move and contain the nuclear waste material. We explore the gouging that occurs when stainless steel canisters are lowered into the storage vault and how gouging compromises the integrity of the containers. Finally, we examine management practices at San Onofre and an apparent lack of supervision, training and protocols. The examination of the perils of S.O.N.G.S. Independent Spent Fuel Storage Installations' poor location, poor technology and poor management, presents an urgent situation for regulators to: order Edison to permanently stop the loading of canisters into dry storage, require Edison to store the waste in canisters that may be inspected, and secure an independent analysis and risk assessment of canister loading procedure.

RATIONALE

Most serious of the issues facing the interim storage of nuclear waste at S.O.N.G.S. include the gouging damage to fully-loaded steel canisters upon downloading into the storage vault. These 54-ton thin-walled steel canisters are loaded with nuclear waste in wet storage – spent fuel pools – and are transported to the on-site concrete storage vault, adjacent to the reactor domes. With the Brinell hardness scale calculations our team demonstrates the depth and width of canister gouges upon downloading into the storage system. The current downloading procedure and on-site storage configuration provides the factors necessary to create gouges in the external steel walls of the canisters: operators have no visibility of the canister during downloading and precise adjustments to canister orientation cannot be made. These gouges remain undetected and unrepaired due to the lack of thorough inspection and monitoring at

the San Onofre Independent Spent Fuel Storage Installations (ISFSIs). The preliminary findings are found in this report.

1. POOR LOCATION

Today, two separate Independent Spent Fuel Storage Installations (ISFSIs) exist at San Onofre. The newest, built by Holtec, is located about 100 feet from the Pacific Ocean on the 85-acre grounds of S.O.N.G.S. The property is part of Marine Corps Base Camp Pendleton and is owned by the Department of the Navy. Two of the nation's busiest transportation corridors -- Interstate 5 and the Los Angeles-San Diego-San Luis Obispo Rail Line -- flank the site. The ISFSIs are clearly visible in Google Earth images and in numerous published photographs. The high accessibility and visibility of the site leaves it extremely vulnerable to an act of malfeasance.

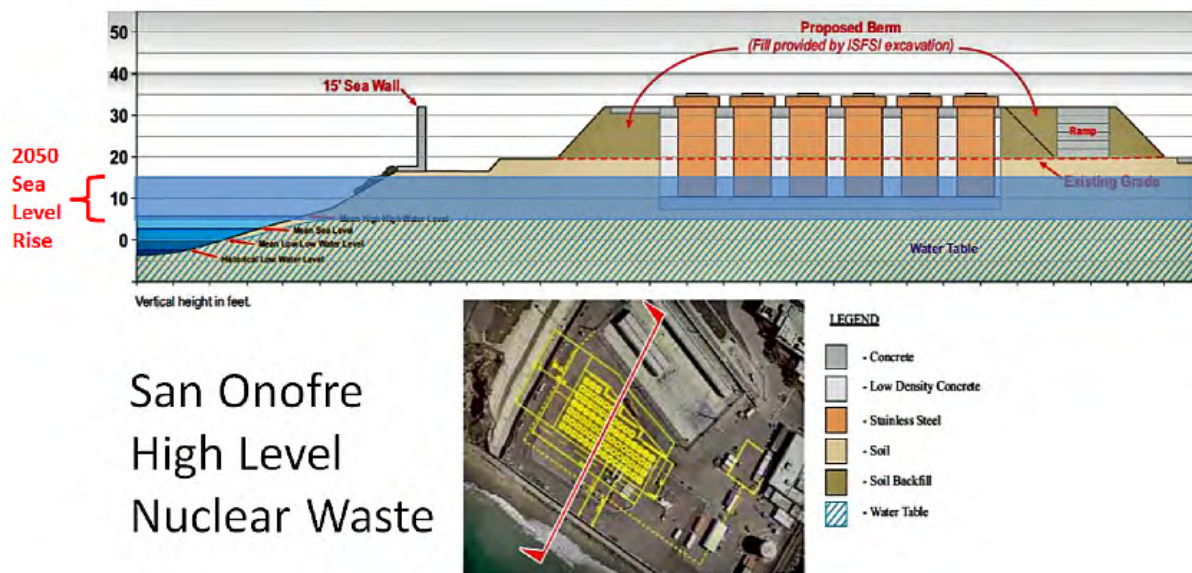


Figure 1. Independent Spent Fuel Storage Installations and Storage Vault.

Forces of nature, exacerbated by sea-level rise, carry further risks. Frequent high humidity and coastal fog make the metal at the site susceptible to short-term corrosion and stress-induced corrosion cracking. Also located at this site is a second, older ISFSI, which contains 51 thin-walled steel canisters that are up to 15 years old.

Numerous reports show that mean high tide level is about 18 inches below the base of the newer, oceanfront ISFSI, which was designed by Holtec. Since this is the mean height, the sea level frequently exceeds this height. Hence, it is likely the present ground water table will leach into the storage vault and result in at least damp storage. Further sea level rise due to climate change will make this problem far worse.

Dr. James Hansen, who managed NASA's climate change program for about 25 years, predicts sea levels could rise up to 10 feet during the next 50 years. At San Onofre, this would cause the bottom seven feet of the Holtec nuclear storage canisters to be submerged in seawater, unintentionally resulting in wet storage. This would invite a crisis similar to that of Fukushima, where spent fuel was exposed to moisture.

A second estimate appears in a comprehensive report by the Working Group of the California Ocean Protection Council Science Advisory Team. Published in 2017, the report shows 75% likelihood sea levels will rise by two feet by 2100. Either of these scenarios envisions that a major portion of the nuclear storage canisters at San Onofre would be submerged in seawater. The combination of the effects of sea-level rise and ground water inundation at the current location would change the Holtec ISFSI to wet storage site, for which it was not designed. Hence, little if anything would be accomplished by moving the waste from the spent-fuel pool to the dry storage ISFSI. The dangers would not be decreased. If anything, the inability to adequately measure and mitigate the impacts of corrosion on the underground nuclear canisters would lead to a significant increase in risk.

All of this can be avoided. If the nuclear waste at the two ISFSIs is transferred into thick-walled casks and then moved to a technically defensible storage facility at higher ground, the problems of ocean water and ground water intrusion can be avoided. As an added benefit, the waste would be easier to secure from an act of malfeasance.

2. POOR TECHNOLOGY

In California, the storage tanks at gas stations must be double-walled; painful experience has shown that single-walled containers can leak gasoline into the groundwater system. With a double-walled fuel tank, if a leak occurs it can be detected and the storage container can be repaired or replaced before any gasoline is released. At San Onofre, we certainly should expect that some kind of leak prevention system would be in place to contain extremely toxic high-level radioactive waste. Additionally, the canisters should be able to be monitored and inspected. The thin-walled canisters at the San Onofre ISFSIs cannot be adequately monitored or inspected. Regulators and Holtec officials have stated that the canisters cannot be inspected from the inside or the outside for cracks or other degradation and that, even if damage could be identified, it would be impossible to fix.

To illustrate the importance of adequate monitoring, we analyze a scenario in which one vent of a canister clogs. We refer to a Holtec non-proprietary safety analysis report¹ that calculates a temperature rise to about 90% of the maximum permissible limit (MPL) in 24 hours. This infers that within the next 12 hours the system will exceed the MPL rating and lead to a meltdown².

¹ Table 4.I.9, page 1050, Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System. USNRC Docket No.: 72-1014, Holtec Report No.: HI-2002444.

² S. Alyokhina, Thermal analysis of certain accident conditions of dry spent nuclear fuel storage, Nuclear Engineering and Technology 50 (2018) 717-723.

Through our own statistical analysis,³ we prove that if the probability of clogging one of the vents during an event is 1%, then the chance that one of the 146 total vents (two vents on each of 73 canisters) will clog in such an event is 78%. This chance reduces to 53% if we reduce the probability of occurrence to .5% from 1%. Tsunamis followed by clogging are dependent events and thus the combined chance of such an event is about 11% during a 30-year period. The sea level rise, the rise of tide levels and the associated rise in the coastal aquifer are all interlinked, as discussed previously. These climate-related phenomena could cause serious damage to the ISFSIs. Therefore, close monitoring and the use of proven thick-walled cask technology for all nuclear waste storage containers is not only necessary but urgent. A mishap could imperil the lives and livelihoods of more than 8 million people who live within 50 miles of the ISFSIs.

2.1 NEAR MISS EVENT

David Fritch, an industrial safety inspector turned whistleblower, remembers August 3, 2018, as a bad day. Fritch worked at San Onofre during a loading failure that left a fully-loaded 54-ton canister of high-level radioactive waste stuck on the lip of a guide ring. Above the 17-foot-tall canister, the slings that attached it to the behemoth loading rig had gone slack.

The canister was, “hanging by about a quarter inch,” Fritch told attendees of the community engagement panel on August 9. “It’s a bad day. That happened, and you haven’t heard about it, and that’s not right. What we have is a canister that could have fallen 18 feet.”

Subsequent investigations revealed that the operators and managers could not see Canister No. 29 as it was being loaded into the storage cavity and became stuck for nearly an hour.

Since the near-accident, regulators have halted further loading of canisters into the seaside storage vault and researchers have explored what could have happened if Canister No. 29 had fallen.

Our own research explores the basic physics of a fully-loaded 54-ton canister in free fall to extrapolate the upper energy involved in the initial impact.

For example, the falling canister could hit the steel-lined concrete floor of the nuclear waste storage facility with explosive energy greater than that of several large sticks of dynamite. The resultant damage to the canister could cause a large radiation release.

At point of contact at the bottom of the storage cavity, damage to the concrete and metal structure could ruin the cooling system. The damage to the concrete would equal that of a fully-loaded 18-wheeler truck, with a gross weight of 80,000 pounds, crashing into reinforced concrete at 23 miles per hour. Our preliminary calculations show the combination of the weight and velocity of the dropped canister exceeds the ISFSIs’ “design criteria for tornado missiles,” by a factor of 4. Future experiments should include drop tests of the actual canisters with non-

³ Chakraborty and English, 2019, ES&H Risk Estimation from “Interim Storage” of SNF at the Beach: The San Onofre NPP, WM2019 Conference, March 3-7, 2019, Phoenix, Arizona, USA (under review).

radioactive loads that simulate the weight of the spent fuel assemblies and fuel baskets to determine what would happen to the actual canisters.

Southern California Edison is set to move 73 canisters into the seaside storage vault and, at the time of publication, has moved 29. Each nuclear storage canister contains 37 spent fuel assemblies, which generate enormous amounts of heat. The systems are cooled by a simple air duct system, which could have been blocked by the damage caused by the canister's fall. If that had happened, great quantities of water would have been needed to cool the reaction and prevent or control a meltdown. The enveloping water would instantly become radioactive steam, as we saw at Fukushima. In the heavily-populated area surrounding San Onofre, however, radioactive steam could prompt the evacuation of millions of people. What's more, since both the canister and the surrounding structure could be badly damaged, there would be no available way to pull the damaged canister from the storage cavity.

Nuclear Regulatory Commission (NRC) computer simulations show what happens when a nuclear storage canister with slightly thinner walls⁴ drops from 19 feet. In the test, a canister falls from a transfer cask onto a storage pedestal. The canister failure rate was 28%. Similar calculations must be performed at San Onofre to determine if that storage system has a similar probability of canister failure. At 28%, that is more than a one-in-four chance of catastrophic failure. Would you fly on an airplane with those odds? Our analysis alone should place the NRC, policymakers and Edison on alert. A more substantial analysis must be completed to examine the potential damage that can be caused by a falling, fully-loaded 54-ton nuclear storage canister.

Continued loading of the nuclear waste into canisters threatens the lives and livelihood of more than 8 million people. Software and computer resources are available by which estimates can be made of the impacts of a dropped canister on both the reinforced concrete and the canister walls. The NRC-approved Holtec technical specifications state that a canister drop of more than 11 inches requires the contents of the canister to be inspected for damage. This specification assumed the canister was in a transfer cask. The impact of an un-casked canister was never analyzed because Holtec and the NRC assumed it could never happen, citing triple-redundancy of the fuel transfer system. But a subsequent NRC inspection revealed that on August 3rd, all three components of this system simultaneously failed. Only the accidental snag of a quarter-inch of the 54-ton canister on the lip of the guide ring prevented a catastrophe.

Our research suggests the entire storage system may need to be redesigned to reduce the probability of canister failure to levels that are acceptable in such a highly-populated area.

⁴ Pg. 4-24 Table 12, NUREG-1864 - A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant, March 2007, A. Malliakos, NRC Project Manager

RESULTS

2.2 GOUGES IN DROPPED CANISTER

In their 2007 report, the NRC's analysts did not consider the impact of gouges on the strength of canister walls. There was no need, the analysts and a Holtec official said, as gouges were not important to the system under examination. We disagree. A detailed analysis of gouging is necessary to properly evaluate the damage to Canister No. 29 during the botched loading and to every other canister loaded into the ISFSI.

We established preliminary results of such an analysis using the Brinell hardness scale approach to estimate the depth and width of expected gouges in 316 stainless steel, of which the Holtec canisters at San Onofre is made.

While the canister is stuck, the guide ring gouges the bottom of the canister.

As the canister drops it is gouged on two sides by a combination of the guide ring, the storage cavity wall and the inner diameter of the transfer cask. This gouging absorbs some of the kinetic energy of the canister.

When the canister smashes into the bottom of the cavity, the kinetic energy and momentum from the fall will be dissipated by damage to:

- the ISFSI;
- the canister; and
- the contents of the canister.

The formation process of gouges will exert a force on the canister. This is the force, P , shown in Figure 2.

Brinell Hardness Scale

The **Brinell scale** characterizes the indentation hardness of materials through the scale of penetration of an indenter, loaded on a material test-piece.

$$\text{BHN} = \frac{2P}{\pi D(D - \sqrt{D^2 - d^2})}$$

Where:

BHN = Brinell Hardness Number (kgf/mm²)
 P = applied load in kilogram-force (kgf)
 D = diameter of indenter (mm)
 d = diameter of indentation (mm)

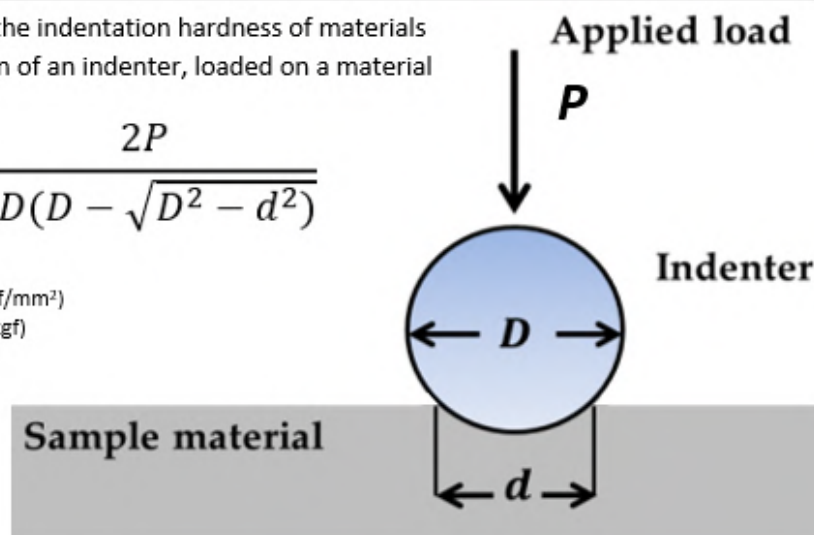


Figure 2. Brinell hardness scale calculation. Credit: *The Samuel Lawrence Foundation.*

In Figure 3, the width of a gouge is shown in relationship to the canister's weight. The expected range of gouge widths is shown in Figure 3. A variety of indenter widths are used as a surrogate for the gouging. The gouging widths range from 2 mm to 16 mm. This is highly significant, since the thickness of the nuclear canisters is 5/8", which is close to 16 mm. We recommend that tests be performed on actual canisters to experimentally determine the accuracy of these predictions.

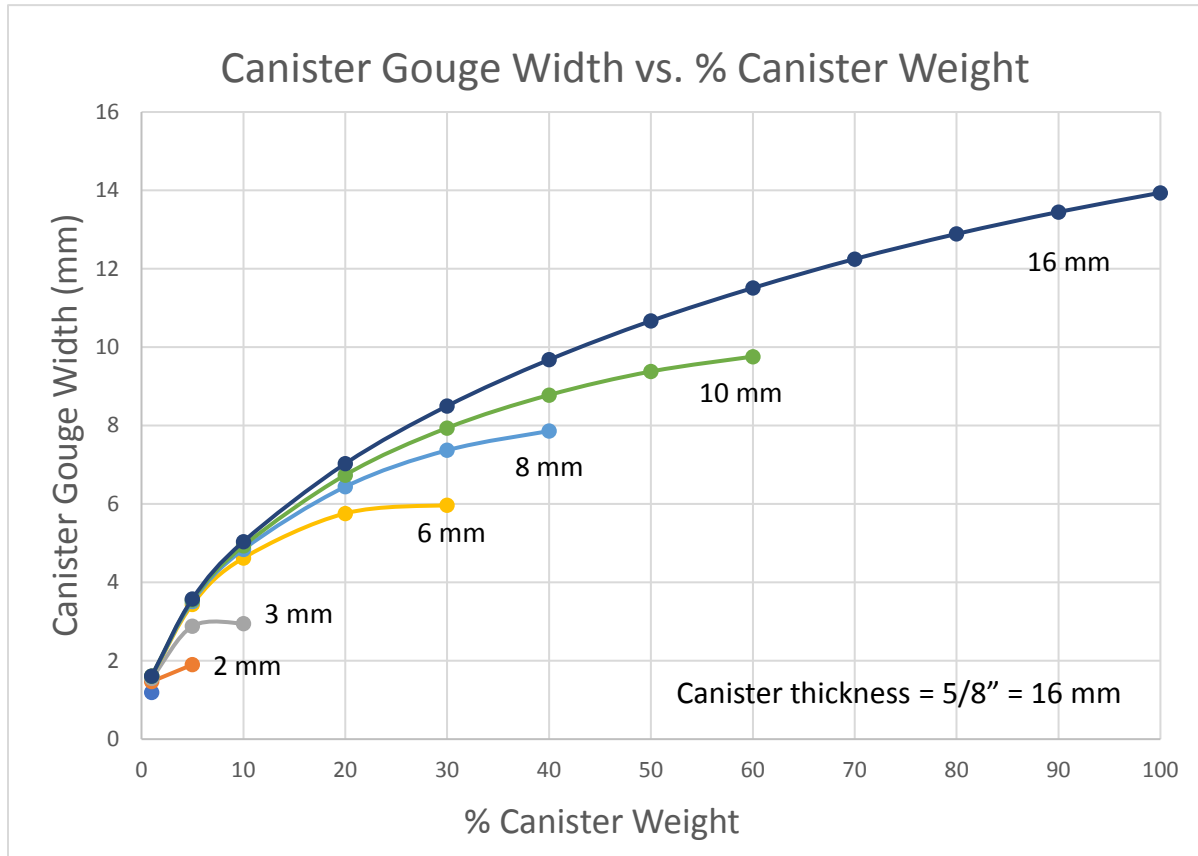


Figure 3. Calculated penetration width of gouge as a function of load for different indenter diameter. The hardness number in Brinell scale for stainless steel 316 (BHN) is 217 kgf/mm². Saturated zone is eliminated.

The expected range of gouge depths is shown in Figure 4. A variety of indenter depths are used as a surrogate for the gouging. The gouging depths expected to be found range from 1 mm to 4.5 mm. This is highly significant, since 4.5 mm is 28% of the thickness of the nuclear storage canister.

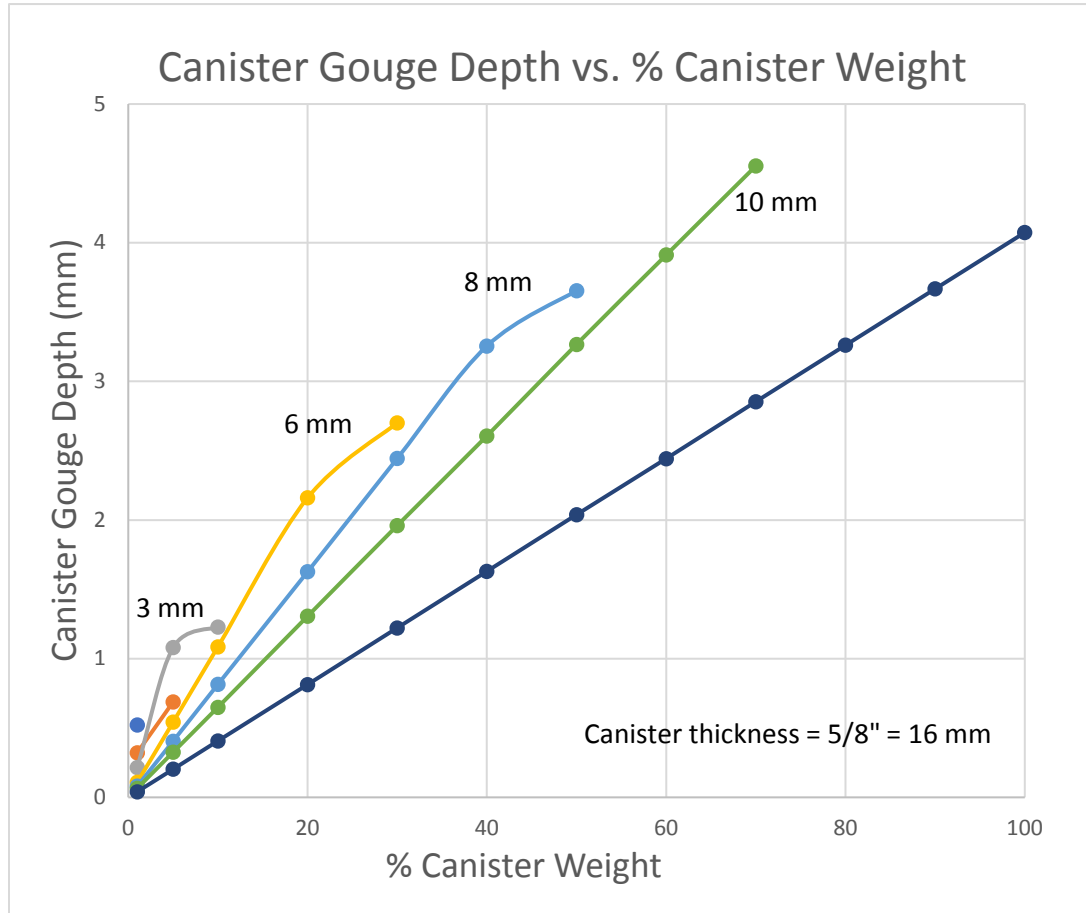


Figure 4. Calculated penetration depth of gouge as a function of load for different indenter diameter. The hardness number in Brinell scale for stainless steel 316 (BHN) is 217 kgf/mm².

2.3 GOUGES DURING ROUTINE LOADING

Extensive gouging will also occur during routine loading of the nuclear storage canister into the storage cavity. By moving the Vertical Cask Transporter, shown in Figure 5, crude adjustments can be made to the alignment of the canister as it is lowered into the storage cavity. The bulky, tank-like machine travels on steel treads, like those found on earth-moving or military equipment. The transporter is not equipped to make the fine adjustments required to insert the nuclear storage canister into the narrow spacing of the storage cavity without banging the canister against the guide ring. This banging gouges the canister and causes the canister to move side-to-side, similar to a pendulum. An Edison official has referred to this process as "jiggling." This jiggling process continues for 15 to 30 minutes as the canister is lowered to the bottom of the storage cavity. Each "jiggle" causes the type of gouging shown in Figure 3 and

Figure 4. We expect that this routine loading process produces a multitude of gouges that significantly damage the canister walls, rendering them unsuitable for storage of nuclear waste.



Figure 5. Vertical Cask Transporter during downloading and alignment of a canister.

Credit: San Onofre Special Inspection Webinar Presentation (NRC).

We strongly recommend that a sampling of the canisters previously lowered into the storage vault be removed and inspected so the extent of gouging can be experimentally determined. We expect the damage will be so severe that the current ISFSI will need to be replaced.

3. POOR MANAGEMENT

During the late 1970s and early 1980s, Rear Admiral Len Hering, USN (ret) served as a Nuclear Weapons Safety Officer, Handling Officer and Surety Officer. Admiral Hering provides the following assessment of management practices at the S.O.N.G.S. ISFSI.

When it comes to the handling and movement of nuclear material, you would expect that only those specifically qualified and trained for such an important task would be deployed to ensure the safe movement of that material. In the Department of Defense (DOD), strict requirements are in place to make sure this very dangerous material is properly handled, transported and stowed.

The DOD and Navy programs were created and built to make certain nuclear material was secure, safely handled and accounted for. Every person who has any contact with nuclear material is required to have a security clearance. A “two-person rule” is in effect at all times. Personnel at all levels perform countless hours of training, obtain certifications of qualification, and complete rigorous inspection and training events to both prove and assure their proficiency in performing the job they are assigned. All of this is all done before anyone is permitted to even gaze upon a real weapon.

Handling gear and all aspects of the evolution are vigilantly maintained, inspected, weight-tested and inspected again. Cranes and dollies or hoist equipment are tested, placed under extreme loading conditions and prepared for specific tasks. Nothing goes untested. Nothing. We leave nothing to chance and we never hypothetically presume. If it isn’t tested and proven, it isn’t done with the actual material in question.

Ashore, and specifically at S.O.N.G.S, I find that virtually none of the protocols that should be expected for the safe handling of this dangerous material are present. I find that personnel and companies are being hired virtually off the street, no specific qualification standards are present or for that matter even required, training is not specific to the risks of the material involved, and there is no fully-qualified and certified team assembled for this highly-critical operation. They have not been required to conduct dry runs to ensure handling teams are proficient and, more importantly, they have never trained specifically to be ready to execute emergency procedures should the unexpected occur. The manuals are not on site, nor are they being followed to step a team through the evolution of moving the nuclear waste. Team leaders have no specific handling qualifications or training. Even the industrial safety inspectors are not specifically nuclear-certified but are general industrial specialists. No manuals are available for procedural review and, by their own admission, the required number of safety officials are often absent during movement of the nuclear storage canisters. In the Navy, if a near-accident such as the one at S.O.N.G.S is uncovered, the Commanding Officer, Weapons Officer -- and anyone else with a significant position on the team -- are relieved. The ship is then ordered to stand-down while a team of experts off-loads its cargo.

The widely reported incident in which a 54-ton, thin-walled container nearly fell 18 feet while it was being lowered into its silo rocked me to the core. What made things worse was narrative in a follow-up report that stated the canister was left suspended for nearly an hour, held up by a mere guide ring installed in the silo, cables slack and operators clueless. There is no doubt that this incident occurred because those on-scene were completely unqualified, unprepared, untrained and incompetent. This very dangerous operation was being performed as if this crew were moving a simple stack of wood around a construction site when, in actuality, the crew was conducting one of the most dangerous operations in the industrial sector. No one was relieved, fired or held accountable. The investigation being conducted is flawed in that those responsible for this deplorable safety environment are the same people who will feed findings to the investigation.

The handling of nuclear waste at San Onofre and other sites across our country should scare every single American. We have a regulatory agency that has failed to make sure the most basic safety precautions are being applied to one of the most dangerous industrial evolutions of our time. The number of waivers being issued where safety is of concern is staggering.

In the DOD, the reason why there were and continue to be no significant accidents with the handling of nuclear material is because there are no waivers and there are no quick wins. Workers are fully qualified, inspected and certified to handle this very dangerous material. In this case, there is no room for error. One mistake is too many. It is my professional opinion that we need to hit the reset button before a disaster of unparalleled portion occurs.

CONCLUSION

The nuclear waste at San Onofre requires a much better storage configuration and must be moved to a technically defensible storage facility to reduce threats. From a security standpoint, the waste should be moved further away from major transportation corridors. The thin-walled nuclear waste storage canisters are at risk of failure due to gouging when downloaded into the seaside storage vault. Once lowered into the storage system, the canisters cannot be thoroughly inspected, monitored or repaired. A near-accident on August 3rd demonstrated that safety protocols are lacking, and that further study is needed to understand the consequences of dropping a fully-loaded 54-ton canister of nuclear waste. The incident revealed that the loading equipment is imprecise and revealed a pattern of mismanagement in canister loading procedure. A complete analysis of canister loading procedure and comprehensive risk assessment must be conducted by an independent party with absolute transparency. If an accident, natural disaster, negligence, or an act of terrorism were to cause a large-scale release of radiation, the health and safety of 8.4 million people within a 50-mile radius would be put at risk. To secure the nuclear waste properly, we recommend a permanent stop to the loading of nuclear storage canisters into the seaside storage vault, placing spent fuel into reliable canisters that can be monitored, inspected and repaired, and moving these canisters to an acceptable storage facility at a significantly higher elevation.

ACKNOWLEDGEMENTS

We thank UCSD Departments of Chemistry and Biochemistry and The Samuel Lawrence Foundation. For more information visit www.samuellawrencefoundation.org/nuclear-energy.

EXHIBIT 7

EXHIBIT 28

ARTICLE XII. CONTRACTOR'S WARRANTIES

12.1 WARRANTIES.

(a) Contractor warrants to Company that all Equipment shall be (i) new and of good quality; (ii) free from improper workmanship and Defects; (iii) conform to all applicable requirements of all Applicable Laws and all Applicable Permits; and (iv) be fit for Company's use in the nuclear power industry for the intended purpose. If Contractor accepts the Existing Canisters for use, Contractor warrants that the Existing Canisters shall be free from Defects or improper workmanship to the extent caused by or due to Contractor's acts or omissions.

(b) Contractor warrants to Company that the Work will be performed in a good and workmanlike manner, and that the Work will: (i) conform to and be designed, engineered and constructed in accordance with the Drawings, Scope of Work, all Applicable Laws and Applicable Permits and other terms of the Contract Documents; (ii) conform with, and be designed and engineered according to professional standards and skill, expertise and diligence of design professionals regularly involved in decommissioning projects similar to the Project, and comply with the requirements of the relevant Government Authorities, including the NRC; (iii) be suitable for the use as set forth in the Technical Specification; (iv) be compatible with the spent fuel pools for Units 2 and 3, spent fuel, fuel handling building, the existing ISFSI, Jobsite, and the SONGS site conditions; (v) contain the Equipment, supplies and materials described in the Scope of Work, all installed in accord with the applicable Contract Documents; (vi) in the case of Apparatus be designed, engineered, licensed, fabricated and manufactured using appropriate and approved processes, procedures and materials and to comply with and satisfy all the terms of the Certificate of Compliance issued by the NRC to Contractor as modified or amended as contemplated herein; (vii) in the case of Drawings or documents required hereunder, accurately and completely present information required to be included therein or necessary to avoid misunderstandings of the included content; and (viii) at such times as the NRC issues or amends a Certificate of Compliance with respect to an Apparatus or Existing Canisters, as applicable, the Apparatus or such Existing Canister specifically approved by the NRC to perform functions required by regulation as described in such Certificate of Compliance shall perform its required functions set forth in such Certificate.

(c) Contractor warrants to Company that all of the documents prepared by Contractor for submittal to a Government Authority for review and approval shall be prepared in full compliance with Applicable Laws and in form and substance such that Company shall not be

required to modify or revise such documents due to a failure to include any required information, inaccuracies or the use of inappropriate forms or formats.

(d) Contractor warrants to Company that none of the Work, including the Equipment (but not including the Existing Canisters), the Drawings, Final Plans and the design, engineering and other services rendered by Contractor hereunder, nor the use or ownership thereof by Company in accordance with the licenses granted hereunder, infringes, violates or constitutes a misappropriation of any trade secrets, proprietary rights, intellectual property rights, patents, copyrights or trademarks.

(e) Except as expressly stated herein to the contrary, Contractor warrants that it shall remedy, in accordance with Section 12.2, any Defects in the Work due to faulty design, materials or workmanship which appear within a period commencing upon the date of ISFSI Scope Completion and continuing for the applicable period following the ISFSI Scope Completion Date (as such period may be extended in accordance with the terms hereof, the "Warranty Period"), as follows:

- (i) with respect to the MPC-37 canisters, twenty five (25) years;
- (ii) with respect to Contractor's Work on Existing Canisters used to store non-fuel waste from the spent fuel pools, twenty five (25) years; provided that the Warranty Period with respect to such Work shall commence on the date that the last of the Existing Canisters containing non-fuel waste are loaded on the ISFSI during Post-ISFSI Scope Work and the related Milestone has been completed;
- (iii) with respect to the Contractor's Work on Existing Canisters used to store greater than class "C" radioactive waste from reactor vessel segmentation in the Post-ISFSI Scope Work, twenty five (25) years; provided that the Warranty Period with respect to such Work shall commence on the Final Acceptance Date;
- (iv) with respect to the HI-STORM UMAX System, ten (10) years;
- (v) with respect to any other Work that is required to be completed in order to achieve ISFSI Scope Completion, including Contractor's Work on any newly assembled AHSM-HS modules that are used by Contractor in the performance of the Work, two (2) years; and
- (vi) with respect to any other Work that is completed after the ISFSI Scope Completion Date, two (2) years from the Final Acceptance Date.

Contractor shall bear all costs of corrections and repairs during the Warranty Period. The provisions of this Section 12.1 apply to Work performed by Subcontractors as well as Work performed directly by Contractor. The provisions of this Article XII do not apply to corrective work caused by the acts or omissions of Company or any separate contractor of Company. If and in the event Company notifies Contractor of a Defect within the Warranty Period, Contractor, at Contractor's expense, shall perform all Work necessary to remedy the Defect, and the repair or replacement Work performed by Contractor to accomplish that purpose shall be subject to an additional express warranty from the date the repair or replacement is completed which shall continue for a duration equivalent to the original Warranty Period.

(f) Notwithstanding anything to the contrary herein, the warranties set forth in this Section 12.1 shall not apply with respect to any claims to the extent arising from (i) any use of the Work or components thereof by Company that exceeds the requirements or recommendations in Contractor's operation and maintenance manuals; (ii) the failure of any Equipment or Work to be maintained in accordance with Contractor's written instructions; or (iii) the modification of any Equipment or Work without Contractor's written consent.

(g) THE WARRANTIES OF CONTRACTOR SET FORTH IN THIS AGREEMENT ARE EXCLUSIVE AND IN LIEU OF ALL OTHER WARRANTIES, WHETHER STATUTORY, EXPRESS OR IMPLIED (INCLUDING ALL WARRANTIES OF MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE AND ALL WARRANTIES ARISING FROM COURSE OF DEALING AND USAGE OF TRADE). The foregoing sentence is not intended to disclaim any other obligations of Contractor set forth herein.

12.2 REPAIR OF NONCONFORMING WORK.

(a) If any of the Work is found to contain Defects, or Contractor is otherwise in breach of any of the warranties set forth in Section 12.1 within the Warranty Period, Contractor shall at its sole cost and expense and without reimbursement hereunder correct, reperform, repair or replace such Defect or otherwise cure such breach as promptly as practicable upon being given notice thereof. Subject to Section 12.3, Company shall give notice to Contractor within two (2) Business Days of discovery of such Defect. Company shall provide Contractor with reasonable access to the Project in order to perform such corrective Work and the Parties shall schedule such corrections or replacements as necessary so as to minimize disruptions to any on-going activities at SONGS. Contractor shall bear all costs and expenses associated with correcting any Defect or breach of warranty, including necessary disassembly, transportation, reassembly and retesting, as well as reworking, repair or replacement of such Work, disassembly and reassembly of piping, ducts, machinery, Equipment or other Work as necessary to give access to improper, defective or non-conforming Work and correction, removal or repair of any damage to other work or property that arises from the Defect. If Contractor is obligated to repair, replace or renew any Equipment, item or portion of the Work hereunder, Contractor will undertake a technical analysis of the problem and correct the "root cause" unless Contractor can demonstrate to Company's satisfaction that there is not a risk of the reoccurrence of such problem. Contractor's obligations under this Section 12.2 shall not be impaired or otherwise adversely affected by any actual or possible legal obligation or duty of any Subcontractor to Contractor or Company concerning any Defect or breach of warranty.

(b) If (i) Contractor fails to complete or commence with due diligence to complete the correction of any Defect or cure of any breach of warranty as required herein within twenty (20) days after receipt of written request from Company to perform such obligations, or (ii) a Defect cannot be corrected within twenty (20) days and Contractor fails to provide a correction plan within five (5) Business Days after receipt of Company's written request to perform such obligations or thereafter fails to implement the plan with due diligence following Company's approval of the plan, then Company may correct or cause to be corrected such Defect or cure such breach of warranty and Contractor shall be liable for all reasonable costs, charges, and expenses incurred by Company in connection therewith (including reasonable and necessary consultants' fees), and Contractor shall, within fifteen (15) days after request therefore, pay to

Company an amount equal to such reasonable costs, charges, and expenses. Any such request by Company shall be accompanied by proper documentation evidencing such reasonable costs, charges and expenses. Any amounts not paid when due shall accrue interest at the Reference Rate (established as of the first day of the month in which payment is due) from the date due until paid. Company and Contractor agree to treat (and shall cause each of their respective Affiliates to treat) any payment made to Company pursuant to this Section 12.2(b) as an adjustment to the Contract Price unless a final determination (which shall include execution of an Internal Revenue Service Form 870-AD or successor form) provides otherwise.

(c) If, during the Warranty Period, Contractor shall change, repair or replace any major Equipment item or component, Company, in its reasonable discretion and consistent with Applicable Laws or Applicable Permits, may require Contractor to assist Company in conducting any test required by any Applicable Law or Applicable Permit with respect to the affected Equipment; provided, however, in connection with any such test, appropriate allowance with respect to the performance of such Equipment shall be made for the fact that such Equipment may have operated prior thereto. If after running such test, the results indicate Contractor has not fulfilled any of its warranty obligations and there is a degradation in the performance of the Project and such degradation results from the warranty Work performed in accordance with this Article XII, then Contractor shall repair, correct or replace such affected Equipment and assist Company in re-running such test until the results no longer indicate a degradation in the performance of the Project resulting from the warranty Work performed in accordance with this Article XII. If Contractor cannot reasonably correct such degraded warranted performance condition then the Parties shall negotiate an equitable settlement of Company's damages based on the amount and scope of such deficient warranted performance, or if the amount of such deficient warranted performance is considered by Company to be a material breach of the terms of this Agreement, then Company may declare such breach to be a Contractor Event of Default pursuant to Section 15.1.

12.3 REPAIRS AND TESTING BY COMPANY.

During the Warranty Period, in the event of an emergency and if, in the reasonable judgment of Company, the delay that would result from giving notice to Contractor could cause serious loss or damage which could be prevented by immediate action, any action (including correction of Defects) may be taken by Company or a third party chosen by Company. Company shall give notice to Contractor within two (2) Business Days of discovery, and in the case of a Defect, the reasonable cost of correction shall be paid by Contractor. In the event such action is taken by Company, Contractor shall promptly respond within five (5) Business Days after correction efforts are implemented, and shall assist whenever and wherever possible in making the necessary corrections. All such warranties obtained shall be in addition to, and shall not alter the warranties of, Contractor. Upon Company's request, Contractor shall use all reasonable efforts to cause Subcontractors to honor warranties including filing suit to enforce same.

12.4 SUBCONTRACTORS. Contractor shall, for the protection of Contractor and Company, obtain from the Subcontractors such guarantees and warranties with respect to Work performed and Equipment supplied, used and installed hereunder as are reasonably obtainable, which guarantees and warranties shall equal or exceed those set forth in Section 12.1 and shall be made available and assignable to Company to the full extent of the terms thereof upon the expiration of Contractor's warranty hereunder. Company shall be an express third party

beneficiary of all such guarantees and warranties, provided such third party beneficiary rights shall not be effective unless this Agreement has been terminated. If available, Company may require Contractor to secure additional warranty or extended guarantee protection pursuant to a Change Order issued in accordance with the provisions of Article VI. Upon the earlier of the ISFSI Scope Completion Date or termination of this Agreement, Contractor shall deliver to Company copies of all relevant contracts providing for such guarantees and warranties.

12.5 CONDITIONS OF WARRANTIES. The warranties set forth in this Article XII are subject to the following conditions applicable to the item for which Company claims a breach of warranty exists:

(a) Company shall notify Contractor in writing of any Defect in the Work as soon as reasonably practicable after Company becomes aware of such Defect.

(b) Company shall have the right to continue to use the Equipment, including the Apparatus, as applicable, or any part thereof, which may require warranty correction or repair until such time as Company elects to remove such Equipment, or part thereof, as applicable, from service; provided, however, in such event, Company shall release Contractor from any additional claims for further defects or damage incurred as a result of such continued operation.

(c) Company shall use and maintain the Equipment, including the Apparatus, in accordance with the operation and maintenance procedures agreed upon by the Parties pursuant to this Agreement (these procedures shall be written by Contractor as part of Contractor's Work so as to integrate (where applicable) or replace and supersede (where not applicable) the operations and maintenance procedures required by the original manufacturer for the Existing Equipment and Existing Canisters such that Contractor may not assert that Company's failure to comply with any separate requirements from the existing manufacturer limits the warranty provided herein by Contractor).

(d) Completion of payments by Company shall not relieve Contractor of any of its warranty obligations.

12.6 ASSIGNMENT OF WARRANTIES. Contractor shall assign to Company or obtain for Company's benefit the manufacturer's warranties for all of the Equipment, including the Apparatus and other deliverables, which are provided in connection with the Work, but which are not manufactured by Contractor, including for Work performed under Section 12.3. Such assignment of warranties to Company must also allow Company to further assign such warranties.

12.7 SURVIVAL OF WARRANTIES. The provisions of this Article XII shall survive the expiration or termination of this Agreement.

EXHIBIT 8

**CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS**

Page 1 of 4

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1040	TBD	TBD	72-1040	0		USA/72-1040

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International
Final Safety Analysis Report for the
HI-STORM UMAX Canister Storage System

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions specified below:

APPROVED SPENT FUEL STORAGE CASK

Model No.: HI-STORM UMAX Canister Storage System

DESCRIPTION:

The HI-STORM UMAX Canister Storage System consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) underground Vertical Ventilated Modules (VVMs), which contains the MPCs during storage; and (3) a transfer cask (HI-TRAC VW), which contains the MPC during loading, unloading and transfer operations. The MPC stores up to 37 pressurized water reactor fuel assemblies or up to 89 boiling water reactor fuel assemblies.

The HI-STORM UMAX Canister Storage System is certified as described in the "UMAX" Final Safety Analysis Report (FSAR) supplemented by the information on the MPCs and transfer cask in the HI-STORM FW FSAR, and in the U. S. Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance (CoC).

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with spent fuel pool water or the ambient environment are made entirely of stainless steel or passivated aluminum/aluminum alloys. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. All confinement boundary components are made entirely of stainless steel. The honeycombed basket provides criticality control.

**CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS**

Supplemental Sheet

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DESCRIPTION (continued)

There are two types of MPCs permitted for storage in HI-STORM UMAX VVM: the MPC-37 and MPC-89. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. Both MPC models have the same external diameter.

The HI-TRAC VW transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the cask loading area to the VVM. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a neutron shield jacket attached to the exterior and a retractable bottom lid used during transfer operations.

The HI-STORM UMAX VVM utilizes a storage design identified as an air-cooled vault or caisson. The HI-STORM UMAX VVM relies on vertical ventilation instead of conduction through the fill material around the VVM, as it is essentially a below-grade storage cavity. Air inlets and an air outlet allow air to circulate naturally through the cavity to cool the MPC inside. The subterranean steel structure is seal welded to prevent ingress of any groundwater in the MPC storage cavity from the surrounding subgrade, and it is mounted on a stiff foundation. The surrounding subgrade and a top surface pad provide significant radiation shielding. A loaded MPC is stored within the HI-STORM UMAX VVM in a vertical orientation.

CONDITIONS**1. OPERATING PROCEDURES**

Written operating procedures shall be prepared for handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 9 of the FSAR.

2. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 10 of the FSAR. At completion of welding the MPC shell to baseplate, an MPC confinement weld helium leak test shall be performed using a helium mass spectrometer. This test shall include the base metals of the MPC shell and baseplate. A helium leakage test shall also be performed on the base metal of the fabricated MPC lid. The confinement boundary welds leakage rate test shall be performed in accordance with ANSI N14.5 to "leaktight" criterion. If a leakage rate exceeding the acceptance criteria is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

3. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important-to-safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the storage system

**CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS**

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Supplemental Sheet

4. HEAVY LOADS REQUIREMENTS

Each lift of an MPC or a HI-TRAC VW transfer cask must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific review of the heavy load handling procedures (under 10 CFR 50.59 or 10 CFR 72.48, as applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.2 of Appendix A.

5. APPROVED CONTENTS

Contents of the HI-STORM UMAX Canister Storage System must meet the fuel specifications given in Appendix B to this certificate.

6. DESIGN FEATURES

Features or characteristics for the site or system must be in accordance with Appendix B to this certificate.

7. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

8. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM UMAX Canister Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

- a. Moving the MPC and the transfer cask into the spent fuel pool or cask loading pool.
- b. Preparation of the HI-STORM UMAX Canister Storage System for fuel loading.
- c. Selection and verification of specific fuel assemblies to ensure type conformance.
- d. Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.
- e. Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool or cask loading pool.
- f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)
- g. Transfer of the MPC from the transfer cask to the VVM.

**CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS**

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- h. HI-STORM UMAX Canister Storage System unloading, including flooding MPC cavity and removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

Any of the above steps can be omitted if the site has already successfully loaded a Holtec MPC System.

9. AUTHORIZATION

The HI-STORM UMAX Canister Storage System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, this certificate, and the attached Appendices A and B. The HI-STORM UMAX Canister Storage System may be fabricated and used in accordance with any approved amendment to CoC No. 1040 listed in 10 CFR 72.214. Each of the licensed HI-STORM UMAX Canister Storage System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

DRAFT

Michele M. Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards
Washington, DC 20555

Dated TBD

Attachments:

1. Appendix A
2. Appendix B

CERTIFICATE OF COMPLIANCE NO. 1040

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM UMAX CANISTER STORAGE SYSTEM

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1.0 USE AND APPLICATION

-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

1.1 Definitions

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AMBIENT TEMPERATURE	AMBIENT TEMPERATURE for Short Term Operations (operations involving use of the HI-TRAC, a Lifting device, and/or an on-site transport device) is defined as the 24 hour average of the local temperature as forecast by the National Weather Service.
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STORM UMAX System are as follows: <ol style="list-style-type: none"> 1. Holtec Generic BWR design 2. Holtec Generic PWR design

1.1 Definitions

<u>Term</u>	<u>Definition</u>
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets, containers or structures that are supporting these loose fuel assembly parts, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.
FUEL BUILDING	The FUEL BUILDING is the site-specific power plant facility, governed by the regulations of 10 CFR Part 50, where the loaded OVERPACK or TRANSFER CASK is transferred to or from the transporter.
GROSSLY BREACHED SPENT FUEL ROD	Spent nuclear fuel rod with a cladding defect that could lead to the release of fuel particulate greater than the average size fuel fragment for that particular assembly. A gross cladding breach may be confirmed by visual examination, through a review of reactor operating records indicating the presence of heavy metal isotopes, or other acceptable inspection means.
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a TRANSFER CASK while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the MPC and end when the TRANSFER CASK is suspended from or secured on the transporter. LOADING OPERATIONS does not include MPC TRANSFER.
MULTI-PURPOSE CANISTER (MPC)	MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
MPC TRANSFER	MPC TRANSFER begins when the MPC is lifted off the TRANSFER CASK bottom lid and ends when the MPC is supported from beneath by the OVERPACK (or the reverse).
NON-FUEL HARDWARE	NON-FUEL HARDWARE is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular

1.1 Definitions

<u>Term</u>	<u>Definition</u>
	Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.
OVERPACK	For the HI-STORM UMAX, the term OVERPACK is synonymous with the term VVM defined below.
PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.
REPAIRED/RECONSTITUTED FUEL ASSEMBLY	Spent nuclear fuel assembly which contains dummy fuel rods that displaces an amount of water greater than or equal to the original fuel rods and/or which contains structural repairs so it can be handled by normal means.
SPENT FUEL STORAGE CASKS (SFSCs)	SFSCs are containers approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM UMAX SFSC System consists of the OVERPACK and its integral MPC.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while an SFSC containing spent fuel is situated within the ISFSI perimeter. STORAGE OPERATIONS does not include MPC TRANSFER.
TRANSFER CASK	TRANSFER CASKs are containers designed to contain the MPC during and after loading of spent fuel assemblies, and prior to and during unloading and to transfer the MPC to or from the OVERPACK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on a TRANSFER CASK loaded with one or more fuel assemblies when it is being moved after LOADING OPERATIONS or before UNLOADING OPERATIONS. TRANSPORT OPERATIONS begin when the TRANSFER CASK is first suspended from or secured on the transporter and end when the TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS includes

1.1 Definitions

<u>Term</u>	<u>Definition</u>
	MPC TRANSFER.
VERTICAL VENTILATED MODULE (VVM)	The VVM is a subterranean type overpack which receives and contains the sealed MPC for interim storage at the ISFSI. The VVM supports the MPC in a vertical orientation and provide gamma and neutron shielding and also provides air flow through cooling passages to promote heat transfer from the MPC to the environs.
UNDAMAGED FUEL ASSEMBLY	UNDAMAGED FUEL ASSEMBLIES are: a) fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial of 3.3 wt% U-235, without known or suspected GROSSLY BREACHED SPENT FUEL RODS, and which can be handled by normal means. An UNDAMAGED FUEL ASSEMBLY may be a REPAIRED/RECONSTITUTED FUEL ASSEMBLY.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC TRANSFER.
ZR	ZR means any zirconium-based fuel cladding or fuel channel material authorized for use in a commercial nuclear power plant reactor.

PURPOSE	<p>The purpose of this section is to explain the meaning of logical connectors.</p> <p>Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.</p>
BACKGROUND	<p>Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.</p> <p>When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.</p>

1.0 USE AND APPLICATION

1.2 Logical Connectors

EXAMPLES The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 VERIFY . . . <u>AND</u> A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . <u>OR</u> A.2.1 Verify . . . <u>AND</u> A.2.2.1 Reduce . . . <u>OR</u> A.2.2.2 Perform . . . <u>OR</u> A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	<p>The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the HI-STORM UMAX System is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the HI-STORM UMAX System is not within the LCO Applicability.</p> <p>Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.</p>

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	<u>AND</u> B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 Completion Times (continued)

EXAMPLES
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit.	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1.	12 hours
	<u>AND</u> B.2 Complete action B.2.	36 hours

When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times (continued)

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

NOTE

Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1.	6 hours
	<u>AND</u> B.2 Complete action B.2.	12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

(continued)

1.3 Completion Times (continued)

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.
---------------------------------	--

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	<p>Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.</p> <p>The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.</p> <p>Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.</p>

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0

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3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	<p>Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5.</p> <p>If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.</p>
LCO 3.0.3	Not applicable.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

- | | |
|----------|---|
| SR 3.0.1 | SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits. |
| SR 3.0.2 | <p>The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.</p> <p>For Frequencies specified as “once,” the above interval extension does not apply. If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.</p> <p>Exceptions to this Specification are stated in the individual Specifications.</p> |
| SR 3.0.3 | <p>If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.</p> <p>If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.</p> |
-

(continued)

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

3.1 SFSC INTEGRITY

3.1.1 Multi-Purpose Canister (MPC)

LCO 3.1.1 The MPC shall be dry and helium filled.

Table 3-1 provides decay heat and burnup limits for forced helium dehydration (FHD) and vacuum drying.

APPLICABILITY: Prior to TRANSPORT OPERATIONS

ACTIONS

NOTES

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure or demohsturizer exit gas temperature limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u> A.2 Develop and initiate corrective actions necessary to return the MPC to compliance with Table 3-1.	30 days

ACTIONS (continued)

B. MPC helium backfill limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential.	72 hours
	<p><u>AND</u></p> <p>B.2.1 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition by adding helium to or removing helium from the MPC.</p> <p><u>OR</u></p> <p>B.2.2 Develop and initiate corrective actions necessary to demonstrate through analysis, using the models and methods from the HI-STORM UMAX FSAR, that all limits for MPC components and contents will be met.</p>	14 days
C. MPC helium leak rate limit for vent and drain port cover plate welds not met.	C.1 Perform an engineering evaluation to determine the impact of increased helium leak rate on heat removal capability and offsite dose.	24 hours
	<p><u>AND</u></p> <p>C.2 Develop and initiate corrective actions necessary to return the MPC to compliance with SR 3.1.1.3.</p>	7 days

D. Required Actions and associated Completion Times not met.	D.1 Remove all fuel assemblies from the SFSC.	30 days
--	---	---------

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify that the MPC cavity has been dried in accordance with the applicable limits in Table 3-1.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill quantity is within the limit specified in Table 3-2 for the applicable MPC model. Re-performance of this surveillance is not required upon successful completion of Action B.2.2.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.3	Verify that the helium leak rate through the MPC vent and drain port cover plates (confinement welds and the base metal) meets the leaktight criteria of ANSI N14.5-1997.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be operable

-----NOTE-----
The SFSC Heat Removal System is operable when 50% or more of the inlet vent duct areas are unblocked and available for flow or when air temperature requirements are met.

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System operable, but partially (<50%) blocked.	A.1 Remove blockage.	N/A
B. SFSC Heat Removal System inoperable.	B.1 Restore SFSC Heat Removal System to operable status.	8 hours
C. Required Action B.1 and associated Completion Time not met.	C.1 Measure SFSC dose rates in accordance with the Radiation Protection Program.	Immediately and once per 12 hours thereafter
	<u>AND</u> C.2.1 Restore SFSC Heat Removal System to operable status.	24 hours
	<u>OR</u> C.2.2 Transfer the MPC into a TRANSFER CASK.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2	Verify all VVM inlets and outlets duct screen are free of blockage from solid debris or floodwater.	24 hours
	<u>OR</u> For VVMs with installed temperature monitoring equipment, verify that the difference between the average VVM air outlet duct temperature and ISFSI ambient temperature is $\leq 80^{\circ}\text{F}$ for VVMs containing MPC-37s and $\leq 85^{\circ}\text{F}$ for VVMs containing MPC-89s.	24 hours

3.1 SFSC INTEGRITY

3.1.3 MPC Cavity Reflooding

LCO 3.1.3 The MPC cavity pressure shall be < 100 psig

-----NOTE-----
The LCO is only applicable to wet UNLOADING OPERATIONS.

APPLICABILITY: UNLOADING OPERATIONS prior to and during re-flooding.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity pressure not within limit.	A.1 Stop re-flooding operations until MPC cavity pressure is within limit.	Immediately
	<u>AND</u> A.2 Ensure MPC vent port is not closed or blocked.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 Ensure via analysis or direct measurement that MPC cavity pressure is within limit.	Once, prior to MPC re-flooding operations.
	<u>OR</u> Once every 1 hour thereafter when using direct measurement.

3.2 SFSC RADIATION PROTECTION.

3.2.1 TRANSFER CASK Surface Contamination.

LCO 3.2.1 Removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC shall each not exceed:

- a. 1000 dpm/100 cm² from beta and gamma sources
- b. 20 dpm/100 cm² from alpha sources.

APPLICABILITY: During TRANSPORT OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each TRANSFER CASK.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TRANSFER CASK or MPC removable surface contamination limits not met.	A.1 Restore removable surface contamination to within limits.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify that the removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC containing fuel is within limits.	Once, prior to TRANSPORT OPERATIONS

3.3 SFSC CRITICALITY CONTROL

3.3.1 Boron Concentration

LCO 3.3.1 The concentration of boron in the water in the MPC shall meet the following limits for the applicable MPC model and the most limiting fuel assembly array/class to be stored in the MPC:

MPC-37: Minimum soluble boron concentration as required by the table below[†].

Array/Class	All Undamaged Fuel Assemblies		One or more Damaged Fuel Assemblies or Fuel Debris	
	Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
All 14x14 and 16x16A	1000	1500	1300	1800
All 15x15 and 17x17	1500	2000	1800	2300

[†] For maximum initial enrichments between 4.0 wt% and 5.0 wt% ^{235}U , the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.0 wt% and 5.0 wt%.

APPLICABILITY: During PWR fuel LOADING OPERATIONS with fuel and water in the MPC

AND

During PWR fuel UNLOADING OPERATIONS with fuel and water in the MPC.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS. <u>AND</u>	Immediately
	A.2 Suspend positive reactivity additions. <u>AND</u>	Immediately
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE-----</p> <p>This surveillance is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC.</p> <p>-----</p>	Once, within 4 hours prior to entering the Applicability of this LCO.
<p>SR 3.3.1.1 Verify boron concentration is within the applicable limit using two independent measurements.</p>	<p><u>AND</u></p> <p>Once per 48 hours thereafter.</p>

Table 3-1
MPC Cavity Drying Limits

Fuel Burnup (MWD/MTU)	MPC Type (Note 5)	Cell Heat Load Limits (Note 6)	Method of Moisture Removal (Notes 1 and 2)
All Assemblies ≤ 45,000	MPC-37 (Short Fuel)	Figure 2.3-1, 2.3-2, or 2.3-3 of Appendix B	VDS (Notes 3 and 4) or FHD (Note 4)
	MPC-37 (Standard Fuel)	Figure 2.3-1, 2.3-2, or 2.3-4 of Appendix B	
	MPC-37 (Long Fuel)	Figure 2.3-5, 2.3-6, or 2.3-7 of Appendix B	
	MPC-89	Figure 2.3-10 of Appendix B	
One or more assemblies > 45,000	MPC-37 (Short, Standard and Long Fuel)	Figure 2.3-12 of Appendix B	VDS (Notes 3 and 4) or FHD (Note 4)
	MPC-89	Figure 2.3-13 of Appendix B	
One or more assemblies > 45,000	MPC-37 (Short Fuel)	Figure 2.3-1, 2.3-2, or 2.3-3 of Appendix B	FHD (Note 4)
	MPC-37 (Standard Fuel)	Figure 2.3-1, 2.3-2, or 2.3-4 of Appendix B	
	MPC-37 (Long Fuel)	Figure 2.3-5, 2.3-6, or 2.3-7 of Appendix B	
	MPC-89	Figure 2.3-10 of Appendix B	

Notes:

1. VDS means a vacuum drying system. The acceptance criterion when using a VDS is the MPC cavity pressure shall be ≤ 3 torr for ≥ 30 minutes while the MPC is isolated from the vacuum pump.
2. FHD means a forced helium dehydration system. The acceptance criterion when using an FHD system is the gas temperature exiting the demister shall be ≤ 21°F for ≥ 30 minutes or the gas dew point exiting the MPC shall be ≤ 22.9°F for ≥ 30 minutes.
3. Vacuum drying of the MPC must be performed with the annular gap between the MPC and the TRANSFER CASK filled with water.
4. Heat load limits are set for each cell; see Appendix B Section 2.3.
5. The fuel assembly lengths loaded in MPC-37 are catalogued as short, standard and long fuel based on the active fuel lengths specified in Appendix B Table 2.1-4.
6. For additional aggregate heat load limits for storage, see Appendix B Table 2.3-1

Table 3-2 MPC Helium Backfill Limits ¹

MPC Type	Helium Backfill Pressure Option	Helium Backfill Pressure Range (psig)
MPC-37	1	≥ 41.0 and ≤ 44.2
	2	≥ 41.0 and ≤ 44.5
	3	≥ 39.0 and ≤ 46.0
MPC-89	1	≥ 42.0 and ≤ 45.2
	2	≥ 39.0 and ≤ 46.0

Note: For Permissible Aggregate Heat Load Limit for each helium backfill pressure option see Appendix B, Table 2.3-1.

¹ Helium used for backfill of MPC shall have a purity of $\geq 99.995\%$. Pressure range is at a reference temperature of 70°F

4.0

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5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

- a. The HI-STORM UMAX Canister Storage System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.
- b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.
- c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

(continued)

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.2 Transport Evaluation Program

- a. For lifting of the loaded MPC or TRANSFER CASK using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.
- b. This program is not applicable when the TRANSFER CASK is in the FUEL BUILDING or is being handled by equipment providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc...).
- c. The TRANSFER CASK when loaded with spent fuel, may be lifted to and carried at any height necessary during TRANSPORT OPERATIONS and MPC TRANSFER, provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.
 1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.
 2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.
 3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program

- 5.3.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.
- 5.3.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.
- 5.3.3 Based on the analysis performed pursuant to Section 5.3.2, the licensee shall establish individual cask surface dose rate limits for the TRANSFER CASK and the VVM to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:
- a. The top of the VVM.
 - b. The side of the TRANSFER CASK
 - c. The outlet vents on the VVM
- 5.3.4 Notwithstanding the limits established in Section 5.3.3, the average of the measured dose rates on a loaded VVM or TRANSFER CASK shall not exceed the following values:
- a. 30 mrem/hr (gamma + neutron) on the top of the closure lid of the VVM
 - b. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK
- 5.3.5 The licensee shall measure the TRANSFER CASK and VVM surface neutron and gamma dose rates as described in Section 5.3.8 for comparison against the limits established in Section 5.3.3 or Section 5.3.4, whichever are lower.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program (continued)

- 5.3.6 If the measured surface dose rates exceed the lower of the two limits established in Section 5.3.3 or Section 5.3.4, the licensee shall:
- Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.
 - Perform a written evaluation to verify whether a VVM at the ISFSI containing the as-loaded MPC will cause the dose limits of 10 CFR 72.104 to be exceeded.
 - Perform a written evaluation within 30 days to determine why the surface dose rate limits were exceeded.
- 5.3.7 If the evaluation performed pursuant to Section 5.3.6 shows that the dose limits of 10 CFR 72.104 will be exceeded, the MPC shall not be placed into a VVM or the MPC shall be removed from the VVM until appropriate corrective action is taken to ensure the dose limits are not exceeded.
- 5.3.8 TRANSFER CASK and VVM surface dose rates shall be measured at approximately the following locations:
- A minimum of four (4) dose rate measurements shall be taken on the top of the VVM. These measurements shall be taken approximately 90 degrees apart around the circumference of the lid, approximately 18 inches radially inward from the edge of the lid.
 - A minimum of four (4) dose rate measurements shall be taken adjacent to the outlet vent duct screen of the VVM, approximately 90 degrees apart.
 - A minimum of four (4) dose rate measurements shall be taken on the side of the TRANSFER CASK approximately at the cask mid-height plane. The measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program (continued)

5.3.9 The "Radiation Protection Space" (RPS) is the prismatic subgrade buffer zone surrounding the VVMs in a loaded ISFSI. The RPS boundary is indicated in the Licensing Drawings in Section 1.5 of the system FSAR. The RPS boundary shall not be encroached upon during any site construction activity. The jurisdictional boundary of the RPS extends down from the top of the ISFSI pad to the elevation of the Bottom surface of the Support Foundation Pad. The ISFSI design shall ensure that there is no significant loss of shielding in the RPS due to a credible accident or an extreme environment event during construction activity involving excavation adjacent to the RPS boundary.

CERTIFICATE OF COMPLIANCE NO. 1040

APPENDIX B

APPROVED CONTENTS AND DESIGN FEATURES

FOR THE HI-STORM UMAX CANISTER STORAGE SYSTEM

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1.0 Definitions

Refer to Appendix A for Definitions.

2.0 APPROVED CONTENTS

2.1 Fuel Specifications and Loading Conditions

2.1.1 Fuel to Be Stored in the HI-STORM UMAX Canister Storage System

- a. UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM UMAX Canister Storage System.
- b. All BWR fuel assemblies may be stored with or without ZR channels.

2.1.2 Fuel Loading

Figures 2.3-1 through 2.3-7 and 2.3-12 define the unique cell numbers for the MPC-37 and MPC-89 models, respectively, and the maximum allowable heat load per fuel assembly for each cell under multiple loading conditions. Fuel assembly decay heat limits are specified in Section 2.3.1. Fuel assemblies shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3.

2.2 Violations

If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:

- 2.2.1 The affected fuel assemblies shall be placed in a safe condition.
- 2.2.2 Within 24 hours, notify the NRC Operations Center.
- 2.2.3 Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

Table 2.1-1 (page 1 of 4)
Fuel Assembly Limits

I. MPC MODEL: MPC-37

A. Allowable Contents

1. Uranium oxide PWR UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and/or FUEL DEBRIS meeting the criteria in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	ZR
b. Maximum Initial Enrichment:	5.0 wt. % U-235 with soluble boron credit per LCO 3.3.1
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling Time \geq 3 years Assembly Average Burnup \leq 68.2 GWD/MTU
d. Decay Heat Per Fuel Storage Location:	As specified in Section 2.3
e. Fuel Assembly Length:	\leq 199.2 inches (nominal design including NON-FUEL HARDWARE and DFC)
f. Fuel Assembly Width:	\leq 8.54 inches (nominal design)
g. Fuel Assembly Weight:	\leq 2050 lbs (including NON-FUEL HARDWARE and DFC)

Table 2.1-1 (page 2 of 4)
Fuel Assembly Limits

I. MPC MODEL: MPC-37 (continued)

- B. Quantity per MPC: 37 FUEL ASSEMBLIES with up to twelve (12) DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS in DAMAGED FUEL CONTAINERS (DFCs). DFCs may be stored in fuel storage locations 1, 3, 4, 8, 9, 15, 23, 29, 30, 34, 35, and 37 (see Figures 2.3-1 through 2.3-7). The remaining fuel storage locations may be filled with PWR UNDAMAGED FUEL ASSEMBLIES meeting the applicable specifications.
- C. One (1) Neutron Source Assembly (NSA) is authorized for loading in the MPC-37.
- D. Up to thirty (30) BRPAs are authorized for loading in the MPC-37.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts, with or without ITTRs, may be stored in any fuel storage location. Fuel assemblies containing APSRs, RCCAs, CEAs, CRAs, or NSAs may only be loaded in fuel storage locations 5 through 7, 10 through 14, 17 through 21, 24 through 28, and 31 through 33 (see Figures 2.3-1 through 2.3-7).

Table 2.1-1 (page 3 of 4)
Fuel Assembly Limits

II. MPC MODEL: MPC-89

A. Allowable Contents

1. Uranium oxide BWR UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and/or FUEL DEBRIS meeting the criteria in Table 2.1-3, with or without channels and meeting the following specifications:

- | | |
|--|--|
| a. Cladding Type: | ZR |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT(Note 1): | As specified in Table 2.1-3 for the applicable fuel assembly array/class. |
| c. Initial Maximum Rod Enrichment | 5.0 wt. % U-235 |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly | |
| i. Array/Class 8x8F | Cooling time \geq 10 years and an assembly average burnup \leq 27.5 GWD/MTU. |
| ii. All Other Array Classes | Cooling Time \geq 3 years and an assembly average burnup \leq 65 GWD/MTU |
| e. Decay Heat Per Assembly | |
| i. Array/Class 8x8F | \leq 183.5 Watts |
| ii. All Other Array Classes | As specified in Section 2.3 |
| f. Fuel Assembly Length | \leq 176.5 inches (nominal design) |
| g. Fuel Assembly Width | \leq 5.95 inches (nominal design) |
| h. Fuel Assembly Weight | \leq 850 lbs, including a DFC as well as a channel |

Table 2.1-1 (page 4 of 4)
Fuel Assembly Limits

II. MPC MODEL: MPC-89 (continued)

B. Quantity per MPC: 89 FUEL ASSEMBLIES with up to sixteen (16) DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS in DAMAGED FUEL CONTAINERS (DFCs). DFCs may be stored in fuel storage locations 1, 3, 4, 10, 11, 19, 29, 39, 51, 61, 71, 79, 80, 86, 87, and 89 (see Figure 2.3-12). The remaining fuel storage locations may be filled with BWR UNDAMAGED FUEL ASSEMBLIES meeting the applicable specifications.

Note 1: The lowest maximum allowable enrichment of any fuel assembly loaded in an MPC-89, based on fuel array class and fuel classification, is the maximum allowable enrichment for the remainder of the assemblies loaded in that MPC.

<p>Table 2.1-2 (page 1 of 3)</p> <p>PWR FUEL ASSEMBLY CHARACTERISTICS</p> <p>(Note 1)</p>					
Fuel Assembly Array/ Class	14x14 A	14x14 B	14x14 C	15x15 B	15x15 C
No. of Fuel Rod Locations	179	179	176	204	204
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.420	≥ 0.417
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3736	≤ 0.3640
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3671	≤ 0.3570
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.563	≤ 0.563
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	5 (Note 2)	21	21
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.015	≥ 0.0165

<p>Table 2.1-2 (page 2 of 3)</p> <p>PWR FUEL ASSEMBLY CHARACTERISTICS</p> <p>(Note 1)</p>					
Fuel Assembly Array/Class	15x15 D	15x15 E	15x15 F	15x15 H	15x15 I
No. of Fuel Rod Locations	208	208	208	208	216
Fuel Clad O.D. (in.)	≥ 0.430	≥ 0.428	≥ 0.428	≥ 0.414	≥ 0.413
Fuel Clad I.D. (in.)	≤ 0.3800	≤ 0.3790	≤ 0.3820	≤ 0.3700	≤ 0.3670
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3735	≤ 0.3707	≤ 0.3742	≤ 0.3622	≤ 0.3600
Fuel Rod Pitch (in.)	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0150	≥ 0.0140	≥ 0.0140	≥ 0.0140	≥ 0.0140

<p>Table 2.1-2 (page 3 of 3)</p> <p>PWR FUEL ASSEMBLY CHARACTERISTICS</p> <p>(Note 1)</p>						
Fuel Assembly Array and Class	16x16 A	17x17A	17x17 B	17x17 C	17x17 D	17x17 E
No. of Fuel Rod Locations	236	264	264	264	264	265
Fuel Clad O.D. (in.)	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377	≥ 0.372	≥ 0.372
Fuel Clad I.D. (in.)	≤ 0.3350	≤ 0.3150	≤ 0.3310	≤ 0.3330	≤ 0.3310	≤ 0.3310
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252	≤ 0.3232	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502	≤ 0.496	≤ 0.496
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 170	≤ 170
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	≥ 0.0350	≥ 0.016	≥ 0.014	≥ 0.020	≥ 0.014	≥ 0.014

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Each guide tube replaces four fuel rods.
3. Annular fuel pellets are allowed in the top and bottom 12" of the active fuel length.
4. One Instrument Tube and eight Guide Bars (Solid ZR)

Table 2.1-3 (page 1 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	7x7 B	8x8 B	8x8 C	8x8 D	8x8 E
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations (Full Length or Total/Full Length)	49	63 or 64	62	60 or 61	59
Fuel Clad O.D. (in.)	≥ 0.5630	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930
Fuel Clad I.D. (in.)	≤ 0.4990	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250
Fuel Pellet Dia. (in.)	≤ 0.4910	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160
Fuel Rod Pitch (in.)	≤ 0.738	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	0	1 or 0	2	1 - 4 (Note 6)	5
Water Rod Thickness (in.)	N/A	≥ 0.034	> 0.00	> 0.00	≥ 0.034
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100

Table 2.1-3 (2 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	8x8F	9x9 A	9x9 B	9x9 C	9x9 D
Maximum Planar-Average Initial Enrichment (wt. % ²³⁵ U) (Note 14)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	64	74/66 (Note 4)	72	80	79
Fuel Clad O.D. (in.)	≥ 0.4576	≥ 0.4400	≥ 0.4330	≥ 0.4230	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3996	≤ 0.3840	≤ 0.3810	≤ 0.3640	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3913	≤ 0.3760	≤ 0.3740	≤ 0.3565	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.609	≤ 0.566	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	N/A (Note 2)	2	1 (Note 5)	1	2
Water Rod Thickness (in.)	≥ 0.0315	> 0.00	> 0.00	≥ 0.020	≥ 0.0300
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.100

Table 2.1-3 (page 3 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	9x9 E (Note 2)	9x9 F (Note 2)	9x9 G	10x10 A	10x10 B
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.5 (Note 12)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	≥0.4170	≥0.4430	≥0.4240	≥0.4040	≥0.3957
Fuel Clad I.D. (in.)	≤0.3640	≤0.3860	≤0.3640	≤ 0.3520	≤ 0.3480
Fuel Pellet Dia. (in.)	≤0.3530	≤0.3745	≤0.3565	≤ 0.3455	≤ 0.3420
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	≥0.0120	≥0.0120	≥0.0320	≥0.0300	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1-3 (page 4 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)			
Fuel Assembly Array and Class	10x10 C	10x10 F	10x10 G
Maximum Planar-Average Initial Enrichment (wt.% ²³⁵ U) (Note 14)	≤ 4.8	≤ 4.7 (Note 13)	≤ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	≥ 0.3780	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3294	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3224	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.488	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	≥ 0.031	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.060

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.
3. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.
4. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
5. Square, replacing nine fuel rods.
6. Variable.
7. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
8. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
9. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
10. These rods may also be sealed at both ends and contain ZR material in lieu of water.
11. Not used.
12. When loading fuel assemblies classified as DAMAGED FUEL, all assemblies in the MPC are limited to 4.0 wt.% U-235.
13. When loading fuel assemblies classified as DAMAGED FUEL, all assemblies in the MPC are limited to 4.6 wt.% U-235.
14. In accordance with the definition of UNDAMAGED FUEL, certain assemblies may be limited to 3.3 wt.% U-235. When loading these fuel assemblies, all assemblies in the MPC are limited to 3.3 wt.% U-235.

Table 2.1-4 CLASSIFICATION OF FUEL ASSEMBLY FOR MPC-37 IN THE HI-STORM UMAX ISFSI		
MPC Type	Classification	Nominal Active Fuel Length
MPC-37	Short Fuel	128 inches \leq L < 144 inches
	Standard Fuel	144 inches \leq L < 168 inches
	Long Fuel	L \geq 168 inches
Note 1: L means "nominal active fuel length".		

2.3 Decay Heat Limits

This section provides the limits on fuel assembly decay heat for storage in the HI-STORM UMAX Canister Storage System. The method to verify compliance, including examples, is provided in Chapter 13 of the HI-STORM UMAX FSAR.

2.3.1 Fuel Loading Decay Heat Limits

Table 2.3-1 provides the maximum permissible decay heat under long-term storage for MPC-37 and MPC-89. Table 2.3-1 also lists the applicable figures providing the permissible decay heat per fuel storage location, including MPCs using the optional helium backfill pressure ranges permitted in Table 3-2 of Appendix A.

TABLE 2.3-1 PERMISSIBLE HEAT LOAD FOR LONG-TERM STORAGE					
MPC Type		Heat Load Chart	Helium Backfill Pressure Option (Notes 1,2)	Permissible Heat Load Per Storage Cell	Permissible Aggregate Heat Load, kW (Note 4)
MPC-37	Short Fuel (Note 3)	1	1	Figure 2.3-1	33.88
		2	2	Figure 2.3-2	33.70
		3	1	Figure 2.3-3	33.53
	Standard Fuel (Note 3)	1	1	Figure 2.3-1	33.88
		2	2	Figure 2.3-2	33.70
		3	1	Figure 2.3-4	35.30
	Long Fuel (Note 3)	1	1	Figure 2.3-5	35.76
		2	2	Figure 2.3-6	35.57
		3	1	Figure 2.3-7	37.06
	Short Fuel (Note 3)		3	Figure 2.3-8	34.28
			3	Figure 2.3-12	33.46
	Standard Fuel (Note 3)		3	Figure 2.3-8	34.28
			3	Figure 2.3-12	33.46
	Long Fuel (Note 3)		3	Figure 2.3-9	36.19
			3	Figure 2.3-12	33.46
MPC-89			1	Figure 2.3-10	36.32
			2	Figure 2.3-11	36.72
			2	Figure 2.3-13	34.75

Notes:

1. For helium backfill pressure option pressure ranges see Appendix A, Table 3-2
2. For the details on the use of VDS to dry High Burnup Fuel see Appendix A, Table 3-1
3. See Table 2.1-4 for fuel length data
4. Aggregate heat load is defined as the sum of heat loads of all stored fuel

assemblies. The permissible aggregate heat load is set to 80% of the design basis heat load.

- 2.3.2 When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any NON-FUEL HARDWARE, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

		1 0.873	2 0.873	3 0.873		
	4 0.873	5 1.602	6 1.602	7 1.602	8 0.873	
9 0.873	10 1.602	11 1.017	12 1.017	13 1.017	14 1.602	15 0.873
16 0.873	17 1.602	18 1.017	19 1.017	20 1.017	21 1.602	22 0.873
23 0.873	24 1.602	25 1.017	26 1.017	27 1.017	28 1.602	29 0.873
	30 0.873	31 1.602	32 1.602	33 1.602	34 0.873	
		35 0.873	36 0.873	37 0.873		

Legend

Cell ID
Heat Load, kW

Figure 2.3-1
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 1 for Long-term Storage for
Short and Standard Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 1.215	2 1.215	3 1.215		
	4 1.215	5 1.080	6 1.080	7 1.080	8 1.215	
9 1.215	10 1.080	11 1.080	12 1.080	13 1.080	14 1.080	15 1.215
16 1.215	17 1.080	18 1.080	19 1.080	20 1.080	21 1.080	22 1.215
23 1.215	24 1.080	25 1.080	26 1.080	27 1.080	28 1.080	29 1.215
	30 1.215	31 1.080	32 1.080	33 1.080	34 1.215	
		35 1.215	36 1.215	37 1.215		

Legend

Cell ID
Heat Load, kW

Figure 2.3-2
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 2 for Long-term Storage for
Short and Standard Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.922	2 0.922	3 0.922		
	4 0.922	5 1.520	6 1.520	7 1.520	8 0.922	
9 0.922	10 1.710	11 0.950	12 0.950	13 0.950	14 1.710	15 0.922
16 0.922	17 1.520	18 0.950	19 0.570	20 0.950	21 1.520	22 0.922
23 0.922	24 1.710	25 0.950	26 0.950	27 0.950	28 1.710	29 0.922
	30 0.922	31 1.520	32 1.520	33 1.520	34 0.922	
		35 0.922	36 0.922	37 0.922		

Legend

Cell ID
Heat Load, kW

Figure 2.3-3
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 3 for Long-term Storage for Short Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.970	2 0.970	3 0.970		
	4 0.970	5 1.600	6 1.600	7 1.600	8 0.970	
9 0.970	10 1.800	11 1.000	12 1.000	13 1.000	14 1.800	15 0.970
16 0.970	17 1.600	18 1.000	19 0.600	20 1.000	21 1.600	22 0.970
23 0.970	24 1.800	25 1.000	26 1.000	27 1.000	28 1.800	29 0.970
	30 0.970	31 1.600	32 1.600	33 1.600	34 0.970	
		35 0.970	36 0.970	37 0.970		

Legend

Cell ID
Heat Load, kW

Figure 2.3-4
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 3 for Long-term Storage for Standard Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.922	2 0.922	3 0.922		
	4 0.922	5 1.691	6 1.691	7 1.691	8 0.922	
9 0.922	10 1.691	11 1.074	12 1.074	13 1.074	14 1.691	15 0.922
16 0.922	17 1.691	18 1.074	19 1.074	20 1.074	21 1.691	22 0.922
23 0.922	24 1.691	25 1.074	26 1.074	27 1.074	28 1.691	29 0.922
	30 0.922	31 1.691	32 1.691	33 1.691	34 0.922	
		35 0.922	36 0.922	37 0.922		

Legend

Cell ID
Heat Load, kW

Figure 2.3-5
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 1 for Long-term Storage for Long Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 1.283	2 1.283	3 1.283		
	4 1.283	5 1.140	6 1.140	7 1.140	8 1.283	
9 1.283	10 1.140	11 1.140	12 1.140	13 1.140	14 1.140	15 1.283
16 1.283	17 1.140	18 1.140	19 1.140	20 1.140	21 1.140	22 1.283
23 1.283	24 1.140	25 1.140	26 1.140	27 1.140	28 1.140	29 1.283
	30 1.283	31 1.140	32 1.140	33 1.140	34 1.283	
		35 1.283	36 1.283	37 1.283		

Legend

Cell ID
Heat Load, kW

Figure 2.3-6
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 2 for Long-term Storage for Long Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 1.019	2 1.019	3 1.019		
	4 1.019	5 1.680	6 1.680	7 1.680	8 1.019	
9 1.019	10 1.890	11 1.050	12 1.050	13 1.050	14 1.890	15 1.019
16 1.019	17 1.680	18 1.050	19 0.630	20 1.050	21 1.680	22 1.019
23 1.019	24 1.890	25 1.050	26 1.050	27 1.050	28 1.890	29 1.019
	30 1.019	31 1.680	32 1.680	33 1.680	34 1.019	
		35 1.019	36 1.019	37 1.019		

Legend

Cell ID
Heat Load, kW

Figure 2.3-7
HI-STORM UMAX MPC-37 Permissible Heat Load Chart 3 for Long-term Storage for Long Fuel

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.785	2 0.785	3 0.785		
	4 0.785	5 1.441	6 1.441	7 1.441	8 0.785	
9 0.785	10 1.441	11 0.915	12 0.915	13 0.915	14 1.441	15 0.785
16 0.785	17 1.441	18 0.915	19 0.915	20 0.915	21 1.441	22 0.785
23 0.785	24 1.441	25 0.915	26 0.915	27 0.915	28 1.441	29 0.785
	30 0.785	31 1.441	32 1.441	33 1.441	34 0.785	
		35 0.785	36 0.785	37 0.785		

Legend

Cell ID
Heat Load, kW

Figure 2.3-8

HI-STORM UMAX MPC-37 Permissible Heat Load for Short and Standard Fuel for Helium Backfill Option 3 in Table 3-2 of Appendix A

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.829	2 0.829	3 0.829		
	4 0.829	5 1.521	6 1.521	7 1.521	8 0.829	
9 0.829	10 1.521	11 0.966	12 0.966	13 0.966	14 1.521	15 0.829
16 0.829	17 1.521	18 0.966	19 0.966	20 0.966	21 1.521	22 0.829
23 0.829	24 1.521	25 0.966	26 0.966	27 0.966	28 1.521	29 0.829
	30 0.829	31 1.521	32 1.521	33 1.521	34 0.829	
		35 0.829	36 0.829	37 0.829		

Legend

Cell ID
Heat Load, kW

Figure 2.3-9
HI-STORM UMAX MPC-37 Permissible Heat Load for Long Fuel for Helium Backfill
Option 3 in Table 3-2 of Appendix A

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

				1 0.431	2 0.431	3 0.431				
		4 0.431	5 0.431	6 0.431	7 0.607	8 0.431	9 0.431	10 0.431		
	11 0.431	12 0.431	13 0.607	14 0.607	15 0.607	16 0.607	17 0.607	18 0.431	19 0.431	
	20 0.431	21 0.607	22 0.607	23 0.607	24 0.607	25 0.607	26 0.607	27 0.607	28 0.431	
29 0.431	30 0.431	31 0.607	32 0.607	33 0.431	34 0.431	35 0.431	36 0.607	37 0.607	38 0.431	39 0.431
40 0.431	41 0.607	42 0.607	43 0.607	44 0.431	45 0.431	46 0.431	47 0.607	48 0.607	49 0.607	50 0.431
51 0.431	52 0.431	53 0.607	54 0.607	55 0.431	56 0.431	57 0.431	58 0.607	59 0.607	60 0.431	61 0.431
	62 0.431	63 0.607	64 0.607	65 0.607	66 0.607	67 0.607	68 0.607	69 0.607	70 0.431	
	71 0.431	72 0.431	73 0.607	74 0.607	75 0.607	76 0.607	77 0.607	78 0.431	79 0.431	
		80 0.431	81 0.431	82 0.431	83 0.607	84 0.431	85 0.431	86 0.431		
				87 0.431	88 0.431	89 0.431				

Legend

Cell ID
Heat Load, kW

Figure 2.3-10

HI-STORM UMAX MPC-89 Permissible Heat Load for Long-Term Storage

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

				1 0.387	2 0.387	3 0.387				
		4 0.387	5 0.387	6 0.387	7 0.546	8 0.387	9 0.387	10 0.387		
	11 0.387	12 0.387	13 0.546	14 0.546	15 0.546	16 0.546	17 0.546	18 0.387	19 0.387	
	20 0.387	21 0.546	22 0.546	23 0.546	24 0.546	25 0.546	26 0.546	27 0.546	28 0.387	
29 0.387	30 0.387	31 0.546	32 0.546	33 0.387	34 0.387	35 0.387	36 0.546	37 0.546	38 0.387	39 0.387
40 0.387	41 0.546	42 0.546	43 0.546	44 0.387	45 0.387	46 0.387	47 0.546	48 0.546	49 0.546	50 0.387
51 0.387	52 0.387	53 0.546	54 0.546	55 0.387	56 0.387	57 0.387	58 0.546	59 0.546	60 0.387	61 0.387
	62 0.387	63 0.546	64 0.546	65 0.546	66 0.546	67 0.546	68 0.546	69 0.546	70 0.387	
	71 0.387	72 0.387	73 0.546	74 0.546	75 0.546	76 0.546	77 0.546	78 0.387	79 0.387	
		80 0.387	81 0.387	82 0.387	83 0.546	84 0.387	85 0.387	86 0.387		
				87 0.387	88 0.387	89 0.387				

Legend

Cell ID
Heat Load, kW

Figure 2.3-11

HI-STORM UMAX MPC-89 Permissible Heat Load for Helium Backfill
Option 2 in Table 3-2 of Appendix A

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

		1 0.97	2 0.97	3 0.97		
	4 0.97	5 0.97	6 0.97	7 0.97	8 0.97	
9 0.97	10 0.97	11 0.7	12 0.7	13 0.7	14 0.97	15 0.97
16 0.97	17 0.97	18 0.7	19 0.7	20 0.7	21 0.97	22 0.97
23 0.97	24 0.97	25 0.7	26 0.7	27 0.7	28 0.97	29 0.97
	30 0.97	31 0.97	32 0.97	33 0.97	34 0.97	
		35 0.97	36 0.97	37 0.97		

Legend

Cell ID
Heat Load, kW

Figure 2.3-12

HI-STORM UMAX MPC-37 Permissible Threshold Heat Load for VDS High Burnup Fuel in Table 3-1 of Appendix A and Helium Backfill Option 3 in Table 3-2 of Appendix A

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

				1 0.44	2 0.44	3 0.44				
		4 0.44	5 0.44	6 0.44	7 0.35	8 0.44	9 0.44	10 0.44		
	11 0.44	12 0.44	13 0.35	14 0.35	15 0.35	16 0.35	17 0.35	18 0.44	19 0.44	
	20 0.44	21 0.35	22 0.35	23 0.35	24 0.35	25 0.35	26 0.35	27 0.35	28 0.44	
29 0.44	30 0.44	31 0.35	32 0.35	33 0.35	34 0.35	35 0.35	36 0.35	37 0.35	38 0.44	39 0.44
40 0.44	41 0.35	42 0.35	43 0.35	44 0.35	45 0.35	46 0.35	47 0.35	48 0.35	49 0.35	50 0.44
51 0.44	52 0.44	53 0.35	54 0.35	55 0.35	56 0.35	57 0.35	58 0.35	59 0.35	60 0.44	61 0.44
	62 0.44	63 0.35	64 0.35	65 0.35	66 0.35	67 0.35	68 0.35	69 0.35	70 0.44	
	71 0.44	72 0.44	73 0.35	74 0.35	75 0.35	76 0.35	77 0.35	78 0.44	79 0.44	
		80 0.44	81 0.44	82 0.44	83 0.35	84 0.44	85 0.44	86 0.44		
				87 0.44	88 0.44	89 0.44				

Figure 2.3-13

HI-STORM UMAX MPC-89 Permissible Threshold Heat Load for VDS
High Burnup Fuel in Table 3-1 of Appendix A and Helium Backfill Option
2 in Table 3-2 of Appendix A

Note that this figure shows the per cell heat load limit for storage. The total permissible aggregate heat load may be less than the sum of each individual cell heat load. See Table 2.3-1 for corresponding permissible aggregate heat load.

3.0 DESIGN FEATURES

3.1 Site

3.1.1 Site Location

The HI-STORM UMAX Canister Storage System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-37

1. Basket cell ID: 8.92 in. (min. nominal)
2. Basket cell wall thickness: 0.57 in. (min. nominal)
3. B₄C in the Metamic-HT: 10.0 wt % (min. nominal)

3.2.2 MPC-89

1. Basket cell ID: 5.99 in. (min. nominal)
2. Basket cell wall thickness: 0.38 in. (min. nominal)
3. B₄C in the Metamic-HT: 10.0 wt % (min. nominal)

3.2.3 Metamic-HT Test Requirements

1. The weight percentage of the boron carbide must be confirmed to be greater than or equal to 10% in each lot of Al/ B₄C powder.
2. The areal density of the B-10 isotope corresponding to the 10% min. weight density in the manufactured Metamic HT panels shall be independently confirmed by the neutron attenuation test method by testing at least one coupon from a randomly selected panel in each lot.
3. If the B- 10 areal density criterion in the tested panel fails to meet the specified minimum, then the manufacturer has the option to reject the entire lot or to test a statistically significant number of panels and perform statistical analysis to show that the minimum areal density in the panels (that comprise the lot) is satisfied with 95% confidence.
4. All test procedures used in demonstrating compliance with the above requirements shall conform to the cask designer's QA program which has been approved by the USNRC under docket number 71-0784.

3.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2007, is the governing Code for the HI-STORM UMAX system MPC as

clarified in Specification 3.3.1 below, except for Code Sections V and IX. However, the HI-STORM UMAX VVM is structurally qualified per the newer 2010 ASME code. The ASME Code paragraphs applicable to the manufacturing of HI-STORM UMAX VVM and transfer cask are listed in Table 3-2. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections, provided a written reconciliation of the later edition against the applicable edition (including addenda) specified above, is performed by the certificate holder. American Concrete Institute ACI-318 (2005) is the governing Code for both plain concrete and reinforced concrete as clarified in Chapter 3 of the Final Safety Analysis Report for the HI-STORM 100 UMAX System.

3.3.1 Alternatives to Codes, Standards, and Criteria

Table 3-1 lists approved alternatives to the ASME Code for the design of the MPCs of the HI-STORM UMAX Canister Storage System.

3.3.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria

Proposed alternatives to the ASME Code, Section III, 2007 Edition, including modifications to the alternatives allowed by Specification 3.3.1 may be used on a case-specific basis when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of the ASME Code, Section III, 2007 Edition, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

(continued)

3.0 DESIGN FEATURES (continued)

<p>TABLE 3-1 List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)</p>			
MPC Enclosure Vessel	Subsection NCA	General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.	<p>Because the MPC is not an ASME Code stamped vessel, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the MPCs as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program.</p> <p>Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.</p>
MPC Enclosure Vessel	NB-1100	Statement of requirements for Code stamping of components.	MPC Enclosure Vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.

<p>TABLE 3-1 List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)</p>			
MPC basket supports and lift lugs	NB-1130	<p>NB-1132.2(d) requires that the first connecting weld of a non-pressure retaining structural attachment to a component shall be considered part of the component unless the weld is more than $2t$ from the pressure retaining portion of the component, where t is the nominal thickness of the pressure retaining material.</p> <p>NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within $2t$ from the pressure retaining portion of the component.</p>	The lugs that are used exclusively for lifting an empty MPC are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The lug-to-Enclosure Vessel Weld is required to meet the stress limits of Reg. Guide 3.61 in lieu of Subsection NB of the Code.
MPC Enclosure Vessel	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC Enclosure Vessel	NB-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are subsumed by the HI-STORM FW FSAR, serving as the Design Specification, which establishes the service conditions and load combinations for the storage system.
MPC Enclosure Vessel	NB-4120	NB-4121.2 and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.	In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, and coating are not, unless explicitly stated by the Code, defined as heat treatment operations.

<p>TABLE 3-1 List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)</p>			
MPC Enclosure Vessel	NB-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.
MPC Enclosure Vessel	NB-4122	Implies that with the exception of studs, bolts, nuts and heat exchanger tubes, CMTRs must be traceable to a specific piece of material in a component.	MPCs are built in lots. Material traceability on raw materials to a heat number and corresponding CMTR is maintained by Holtec through markings on the raw material. Where material is cut or processed, markings are transferred accordingly to assure traceability. As materials are assembled into the lot of MPCs being manufactured, documentation is maintained to identify the heat numbers of materials being used for that item in the multiple MPCs being manufactured under that lot. A specific item within a specific MPC will have a number of heat numbers identified as possibly being used for the item in that particular MPC of which one or more of those heat numbers (and corresponding CMTRS) will have actually been used. All of the heat numbers identified will comply with the requirements for the particular item.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal.

<p>TABLE 3-1 List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)</p>			
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only progressive liquid penetrant (PT) examination is permitted. PT examination will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	<p>The MPC vessel is welded in the field following fuel assembly loading. After the lid to shell weld is completed, the MPC shall then be pressure tested as defined in Chapter 10. Accessibility for leakage inspections precludes a Code compliant pressure test. Since the shell welds of the MPC cannot be checked for leakage during this pressure test, the shop leakage test to 10^{-7} ref cc/sec provides reasonable assurance as to its leak tightness. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination. The MPC lid-to-shell weld shall be verified by progressive PT examination. PT must include the root and final layers and each approximately 3/8 inch of weld depth.</p> <p>The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350.</p>

<p>TABLE 3-1 List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)</p>			
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM UMAX system is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.

Table 3-2 REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS			
	Item	Code Paragraph [2.6.1]	Explanation and Applicability
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The VVM's jurisdictional boundary is defined by the bottom surface of the SFP, the top surface of the ISFSI pad and the SES side surfaces.
3.	Certification of material(structural)	NF-2130(b) and (c)	Materials shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2400	-
6.	Welding procedure	Section IX	-
7.	Welding material	Section II	-
8.	Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3424	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 plate and shell supports and for linear supports are applicable for Closure Lid and Container Shell, respectively.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	Applies to the Container Shell
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Storage of Welding Materials	NF-4411	-
19.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural welds
20.	Backing Strips, Peening	NF-4421	Applies to structural and non-

Table 3-2 REFERENCE ASME CODE PARAGRAPHS FOR VVM PRIMARY LOAD BEARING PARTS			
	Item	Code Paragraph [2.6.1]	Explanation and Applicability
		NF-4422	structural welds
21.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
22.	Non-Destructive Examination	NF-5360	Invokes Section V
23.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	-

3.0 DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

1. The temperature of 80° F is the maximum average yearly temperature.
2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.
3. The resultant zero period acceleration at the top of the grade and at the elevation of the Support Foundation Pad (SFP) at the host site (computed by the Newmark's rule as the sum of $A+0.4*B+0.4*C$, where A, B, C denote the free field ZPA's in the three orthogonal directions in decreasing magnitude, i.e., $A \geq B \geq C$) shall be less than or equal to 1.3 and 1.214, respectively.
4. The analyzed flood condition of 15 fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.
5. The potential for fire and explosion shall be based on site-specific considerations. The user shall demonstrate that the site-specific potential for fire is bounded by the fire conditions analyzed by the Certificate Holder, or an analysis of the site-specific fire considerations shall be performed.
6. The moment and shear capacities of the ISFSI Structures shall meet the structural requirements under the load combinations in Table 3.4-1.
7. Radiation Protection Space (RPS) as defined in Subsection 5.3.9 of Appendix A, is intended to ensure that the subgrade material in and around the lateral space occupied by the VVMs remains essentially intact under all service conditions including during an excavation activity adjacent to the RPS.
8. The SFP for a VVM array established in any one construction campaign shall be of monolithic construction, to the extent practicable, to maximize the physical stability of the underground installation.
9. Excavation activities contiguous to a loaded UMAX ISFSI on the side facing the excavation can occur down to the depth of the bottom surface of the SFP of the loaded ISFSI (i.e. within the area labeled "Space B" in Figure 3-1) considering that there may be minor variations in the depth due to normal construction practices. For excavation activities which are contiguous to the loaded ISFSI (within a distance "W," see Figure 3-1) and below the depth of the bottom surface of the SFP (i.e. within the area labeled "Space D" in Figure 3-1), a site-specific seismic analysis will be performed to demonstrate the stability of the RPS boundary and structural integrity of the ISFSI structure. This analysis shall be submitted to Holtec International to be incorporated in an amendment request for NRC review and approval prior to any excavation taking place.

10. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important-to-safety and must be evaluated to determine the applicable quality assurance category.
11. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area Ambient Temperature $\geq 0^{\circ}$ F.
12. For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any VVM inlet or outlet air ducts for an extended period of time (i.e., longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.
13. Users shall establish procedural and/or mechanical barriers to ensure that during LOADING OPERATIONS and UNLOADING OPERATIONS, either the fuel cladding is covered by water, or the MPC is filled with an inert gas.
14. The entire haul route shall be evaluated to ensure that the route can support the weight of the loaded transfer cask and its conveyance.
15. The loaded transfer cask and its conveyance shall be evaluated to ensure, under the site specific Design Basis Earthquake, that the cask and its conveyance does not tipover or slide off the haul route.

(continued)

DESIGN FEATURES (continued)

Table 3-3 LOAD COMBINATIONS FOR THE TOP SURFACE PAD, ISFSI PAD, AND SUPPORT FOUNDATION PAD PER ACI-318 (2005)	
Load Combination Case	Load Combination
LC-1	1.4D
LC-2	1.2D + 1.6L
LC-3	1.2D + E + L
where: D: Dead Load including long-term differential settlement effects. L: Live Load E: DBE for the Site	

DESIGN FEATURES (continued)

Table 3-4 Values of Principal Design Parameters for the Underground ISFSI
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Thickness of the Support Foundation Pad, inch (nominal)	≥33
Thickness of the ISFSI Pad, inch (nominal)	≥34
Thickness of the Top Surface Pad, inch (nominal)	≥30
Rebar Size* (min.) and Layout* (max)	#11 @ 9" each face, each direction
Rebar Concrete Cover (top and bottom)*, inch	per 7.7.1 of ACI-318 (2005)
Compressive Strength of Concrete at ≤28 days*, psi	≥4500
Compressive Strength of Self-hardening Engineered Subgrade (SES), psi	≥1,000
Lower Bound Shear Wave Velocity in the Subgrade lateral to the VVM (Figure 3-1 Space A), fps**	≥1,300
Depth Averaged Density of subgrade in Space A. (Figure 3-1) ¹	120
Depth Averaged Density of subgrade in Space B. (Figure 3-1) ¹	110
Depth Averaged Density of subgrade in Space C. (Figure 3-1) ²	120
Depth Averaged Density of subgrade in Space D. (Figure 3-1) ³	120
Lower Bound Shear Wave Velocity in the Subgrade below the Support Foundation Pad (Figure 3-1 Space C & D), fps**	≥485
Lower Bound Shear Wave Velocity in the Subgrade laterally surrounding the ISFSI (Figure 3-1 Space B), fps**	≥450

* Applies to Support Foundation Pad and ISFSI Pad.

** Strain compatible effective shear wave velocities shall be computed using the guidance provided in Section 16 of the International Building Code, 2009 Edition. Users must account for potential variability in the subgrade shear wave velocity in accordance with Section 3.7.2 of NUREG-0800.

Notes:

1. A lower average density value may be used in shielding analysis per FSAR Chapter 5 for conservatism.
2. Not required for shielding.
3. This space will typically contain native soil. Not required for shielding.

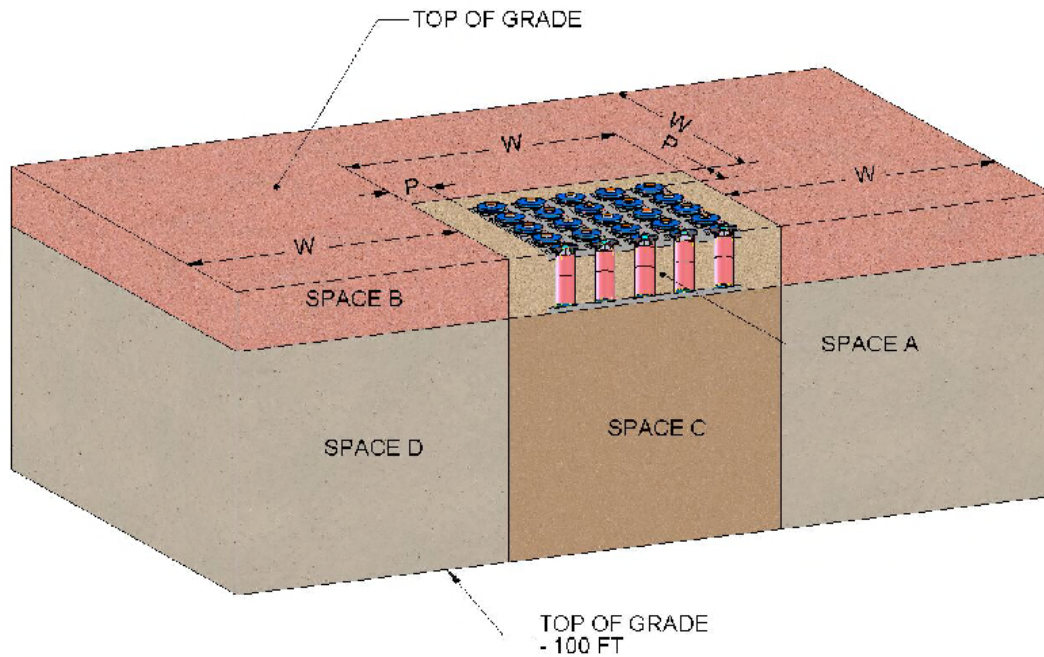


Figure 3-1 - SUBGRADE AND UNDERGRADE SPACE NOMENCLATURE

3.0 DESIGN FEATURES (continued)

3.5 Combustible Gas Monitoring During MPC Lid Welding and Cutting

During MPC lid-to-shell welding and cutting operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present.

3.6 Periodic Corrosion Inspections for Underground Systems

HI-STORM UMAX VVM ISFSIs not employing an impressed current cathodic protection system shall be subject to visual and UT inspection of at least one representative VVM to check for significant corrosion of the CEC Container Shell and Bottom Plate at an interval not to exceed 20 years. The VVM chosen for inspection is not required to be in use or to have previously contained a loaded MPC. The VVM considered to be most vulnerable to corrosion degradation shall be selected for inspection. If significant corrosion is identified, either an evaluation to demonstrate sufficient continued structural integrity (sufficient for at least the remainder of the licensing period) shall be performed or the affected VVM shall be promptly scheduled for repair or decommissioning. Through wall corrosion shall not be permitted without promptly scheduling for repair or decommissioning. Promptness of repair or decommissioning shall be commensurate with the extent of degradation of the VVM but shall not exceed 3 years from the date of inspection.

If the representative VVM is determined to require repair or decommissioning, the next most vulnerable VVM shall be selected for inspection. This inspection process shall conclude when a VVM is found that does not require repair or decommissioning. Since the last VVM inspected is considered more prone to corrosion than the remaining un-inspected VVMs, the last VVM inspected becomes the representative VVM for the remaining VVMs.

Inspections

Visual Inspection: Visual inspection of the inner surfaces of the CEC Container Shell and Bottom Plate for indications of significant or through wall corrosion (i.e., holes).

UT Inspection: The UT inspection or an equivalent method shall be used to measure CEC shell wall thickness to determine the extent of metal loss from corrosion. A minimum of 16 data points shall be obtained, 4 near the top, 4 near the mid-height and 4 near the bottom of the CEC Container Shell all approximately 0, 90, 180, and 270 degrees apart; and 4 on the CEC Bottom Plate near the CEC Container Shell approximately 0, 90, 180, and 270 degrees apart. Locations where visual inspection has identified potentially significant corrosion shall also receive UT inspection. Locations suspected of significant corrosion may receive further UT inspection to determine the extent of corrosion.

Inspection Criteria

General wall thinning exceeding 1/8" in depth and local pitting exceeding 1/4" in depth are conditions of significant corrosion.
