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BY OVERNIGHT MAIL

August 13, 2001

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: USNRC Docket No. 72-1014
HI-STORM 100 Certificate of Compliance 1014
HI-STORM License Amendment Request 1014-1, TAC L23344

References:
1. Holtec Project 5014
2. Holtec Letter, B. Gutherman, to the NRC Document Control Desk, dated July 3, 2001.
3. NRC Letter, C. Jackson to B. Gutherman, dated August 3, 2001

Dear Sir:

We appreciate the SFPO's timely completion of the acceptance review of the above-referenced amendment request. In Attachment 1 to this letter, we provide our responses to the clarifications sought through the enclosure to the August 3, 2001 letter (Reference 3). The changes to LAR 1014-1, Revision 2 required to address these issues will be provided in the form of Supplement 1, to be submitted by August 17, 2001 for arrival at NRC's offices by August 20, 2001.

As you would note from reading Attachment 1, some responses require additional clarifying text to be added to the FSAR and others require certain editorial corrections to be made. We propose to submit revised FSAR pages (labeled Proposed Revision 1C) as part of Supplement 1 to the LAR package. In order to ensure that the proposed revisions to the FSAR in Supplement 1 to LAR 1014-1, Revision 2 have no remaining residual issues from the acceptance review, we are forwarding the draft responses for several of the issues in Attachment 1 herewith. We would appreciate your review of these responses and provide us comments, if any, before the Supplement 1 is packaged on August 17, 2001 for transmittal.

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The responses provided herewith have been reviewed by the licensing committee of the Holtec Users Group. We thank you for your continued attention to our amendment application whose timely review and approval is essential to support our clients' needs.

Sincerely,

Brian Gutherman, P.E.
Licensing Manager

Approval:

K.P. Singh, Ph.D, P.E.
President and CEO

Discipline Concurrence*	
Structural Mechanics Dr. A.I. Soler: Alan I. Soler /mp	Thermal/Hydraulics Dr. Indresh Rampall: Indresh Rampall
Shielding Evaluation Dr. Everett Redmond II:	Civil/Structures Mr. Ray Kellar: Ray Kellar
Operations Mr. Stephen Agace:	
* All Holtec QA-validated submittals on safety significant projects require relevant technical discipline concurrence.	

cc: Mr. Wayne Hodges, USNRC (w/attachment)
Mr. Mark Delligatti, USNRC (w/attachment)
Mr. Christopher Jackson, USNRC (w/attachment)
Mr. Michael Waters, USNRC (w/attachment)

emcc: HUG Group N (w/attachment)
Holtec International Florida Operations (w/attachment)
UST&D, Inc. (Mr. Robert L. Moscardini) (w/attachment)

Document ID: 5014431

Attachment 1: Response to the Issues Raised in Acceptance Review of LAR 1014-1, Revision 2

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U.S. Nuclear Regulatory Commission
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Document I.D.: 5014431
Attachment 1

ACCEPTANCE REVIEW ISSUES ON NEW DESIGN INFORMATION

Response #1:

We agree with the SFPO's suggestion with regard to the proposed FSAR statements; the two sentences will be removed.

Response #2:

As stated in Section 2.0.4.4 of proposed Revision 1B of the HI-STORM 100 FSAR, the jurisdictional interface between the HI-STORM cask system and the ISFSI pad structure is defined by the contact surface between the cask and the ISFSI pad and by the contact surface between the anchor stud's nut and the HI-STORM sector lugs. The anchor studs and the components that constitute the embedment structure are outside the purview of the HI-STORM FSAR. Accordingly, the information provided in the FSAR with respect to interfacing structurals and components, consistent with past practice, is not prescriptive, and the material in Table 2.A.1 is labeled "Representative Materials of Construction."

We agree with the reviewer that carbon steel, in the absence of appropriate corrosion protection measures (such as galvanizing, paint and/or grease to fill spaces where water may intrude and stagnate), would be unsuitable as part of a permanent ISFSI installation. However, if suitably protected against environmental conditions, a carbon steel embedment can remain un-degraded for a long period, as evidenced by the experience of towers and columns installed at chemical and petrochemical plants around the country that have been in service for decades. However, to emphasize the ISFSI designer's responsibility to include environmental considerations in selecting the material for the embedment components, an appropriate footnote will be added to Table 2.A.1.

Response #3

The terminology in Table 2.A.1 will be revised to accord with Figure 2.A.1.

Response #4

To comport with the staff's request, we are adding to proposed FSAR Appendix 2.A the observation that the reinforcing steel (pattern and quantity) may be influenced by the demand of the anchorage forces. Because the structural design of the pad is heavily influenced by the site subgrade conditions, this added text in Appendix 2.A would serve to emphasize the considerations that are already implied through the invoking of ACI-349 for anchored pad designs.

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Response #5

We regret the editorial oversight in Table 3.3.5 of the FSAR: In the revised text (proposed Revision 1C), this error will be corrected.

The minimum density of concrete, previously at 146 and now at 155 pcf, will continue to be a controlled design parameter because the shielding effectiveness of the cask is a direct function of concrete density. Holtec, as we describe in Response #6 below, controls the density of HI-STORM's shielding concrete through a carefully articulated process to ensure that the concrete installed in the HI-STORM overpack maintains a density value at or above the minimum value stated in the FSAR for the design life of the storage system (40 years). The concrete used in the HI-STORM overpacks, however, continues to be referred to as "plain concrete" consistent with the definition implied in ACI 318.1 and ACI 318.1R (revised 1992). The definition of plain concrete will be added to the Glossary of the FSAR to eliminate any ambiguity. The reference to "plain" concrete in the CoC will be proposed to be clarified to say "plain (un-reinforced)" concrete.

Response #6

Holtec controls the density of the concrete placed in the HI-STORM overpacks through a proprietary Holtec procedure (HSP-170). This procedure specifies the required concrete density that must be obtained in the field prior to placement in the overpacks. The field density measurements are obtained in accordance with test methods specified in ASTM C-138, "Standard Test Method for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete." To account for the potential loss of moisture from the concrete over the design life of the overpack, the required minimum wet unit weight (density) of the concrete is specified to be sufficiently heavier than the minimum dry concrete density specified in the FSAR. The basis for the minimum density requirements contained in HSP-170 was established by evaluating the dry weights of the concrete constituents along with the non-evaporable water present in the cement paste from the concrete mix design to ensure that the density remains equal to or greater than the prescribed FSAR minimum density. Thus, while the shielding calculations utilize the minimum dry concrete density set forth in the FSAR; the actual density of the concrete in the overpacks in the field, by virtue of the application of HSP-170, will always be equal to or greater than the prescribed FSAR minimum. Correlational testing of overpack concrete by use of ASTM C-642, "Standard Test Method for Density, Absorption, and Voids in Hardened Concrete" is not required by the applicable ACI codes, nor is it considered necessary based on the methodology for determining concrete density described above.

Response #7

We define "significant" changes as those that bear upon, directly or indirectly, on any technical material contained in the FSAR. Information in the drawings that are so minute as to have no corresponding

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textual material in the FSAR, are considered insignificant. All significant changes are identified in the "List of Effective Pages" in Revision 2 to LAR 1014-1. Based on the staff's request to identify all changes regardless of their significance, a comprehensive list of changes will be prepared for submittal as part of Supplement 1 to LAR 1014-1, Revision 2 to help expedite the ongoing staff reviews.

Response #8

The current licensing basis, as described in FSAR Chapters 2 and 8, recognizes the fact that during every HI-STORM loading evolution, there is a certain length of time during which the loaded MPC is inside the overpack and the overpack lid is not yet in place. This is true for MPC transfers occurring either inside the Part 50 facility or outside the Part 50 facility (i.e., at a Cask Transfer Facility (CTF)).

At a CTF, the loaded overpack must necessarily be moved without the lid installed to allow the HI-TRAC transfer cask to be lowered to the ground. Lid installation is typically performed soon thereafter at a nearby work station, although no specific time limit has been established. Further, no dose analyses or unique accident conditions are described in the FSAR for this transient evolution. The occupational exposures estimated in FSAR Chapter 10 remain applicable because, during this evolution, no additional work activities in a radiation area (e.g., on top of the cask) are required beyond those already evaluated. In other words, for these users, an activity previously presumed to occur after lid installation (i.e., moving the loaded overpack away from the MPC transfer area) now occurs before lid installation.

Moving the overpack from inside the Part 50 facility to outside the Part 50 facility without its permanent lid installed is essentially the same evolution. Differences among users' facilities, such as the truck bay size, cask translocation device, and lid installation facilities may affect the duration of this evolution for a particular user. However, recent experience with HI-STORM cask loadings where MPC transfers took place at a CTF at one site and inside the Part 50 facility at another site indicates that this evolution takes approximately the same amount of time in either scenario.

We believe moving the loaded overpack without the permanent lid installed has already been reviewed and approved by the NRC, based on the following:

1. The movement of the loaded overpack without the lid installed is implicitly recognized in the current, approved FSAR.

2. The NRC's Safety Evaluation Report (Section 8.2) states "All accident conditions applicable to the transfer of the cask to the storage location are bounded by the design events described in Sections 2 and 11 of the SAR."

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The referenced text in Section 8.1.1.2 of Proposed Revision 1B of the FSAR simply adds clarification and defense-in-depth through additional requirements with which users must comply during this evolution. Upon further review of this proposed text in response to this question and discussions with the HUG licensing committee, we recognized two things requiring attention: 1) the proposed text is solely focused on MPC transfers occurring only inside the Part 50 facility and 2) an additional limitation is appropriate. To address these issues, two actions will be taken. First, the requirements in FSAR Section 8.1.1.2 should apply to all cases where the loaded overpack is moved outside the Part 50 facility without its permanent lid installed. Second, a requirement to address the overpack being outdoors without the permanent lid installed during inclement weather needs to be added. The proposed FSAR text will be appropriately modified in Proposed Revision 1C to address both issues.

In conclusion, we note that the short period during which the lid is not situated on the HI-STORM overpack is one scenario out of a multitude that exist during a fuel loading evolution. In view of the varying physical configurations of each plant, the loading process for each site must be developed and evaluated site-specifically. Each of the loading configurations inside the Part 50 jurisdiction is subject to evaluation under 10 CFR 50.59 in accordance with the procedures of the nuclear plant. Part 72 contains a well-articulated set of requirements within §72.212 that require each general licensee to perform written evaluations to ensure the cask system, as deployed at each site, meets the requirements of 10 CFR 72.104, the CoC, and the FSAR. We, therefore, did not find it necessary to re-state the regulations (10CFR72.212) within the body of the FSAR. However, to further reinforce the assertion that Holtec works closely with its users in evaluating deployment of the HI-STORM system, we propose the following verbiage to be inserted in Chapter 8 of the FSAR:

> "The duration for which the HI-STORM overpack is without the top lid shall be minimized and site operating procedures shall specifically address this loading evolution from an ALARA perspective. The distance of HI-STORM movement without the installed lid shall be minimized if the overpack has to be translocated to allow installation of the permanent lid. Evaluations performed by the ISFSI owner to satisfy the provisions of 10CFR 72.212 shall be provided to the CoC holder."

Response #9

The note has been rewritten to remove the ambiguity.

Response #10

We agree with the comment; the second note on page 8.1-18 can cause confusion; it will be deleted from the proposed FSAR changes.

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Response #11

We apologize for the editorial oversight; it will be corrected (in the proposed Revision 1C).

ACCEPTANCE REVIEW ISSUES ON REQUEST FOR ADDITIONAL INFORMATION (RAI)

Response #1

Our response to RAI 4.A.5 was not intended to endorse the CSED approach for SNF burnup in excess of 63.5 GWD/MTU. We cited the CSED method espoused by EPRI as a technical reference in the evolving literature on high burnup SNF, which purports to show that there is a considerable margin in the creep strain limit if the 1% value set forth in ISG-15 is used. While the precise formulation of the CSED method may or may not correlate well with long-term creep data that will inevitably become available in the future, the underlying premise that failure through long-term creep occurs at a larger value of accumulated strain than under a rapidly applied load is a fundamental fact in solid mechanics. Fuketa[1] et al., for example, make this observation in the context of spent nuclear fuel. The modest amount of creep data for irradiated zircaloy that is presently available also suggests that 1% creep limit is extremely conservative for burnups up to 63.5 GWD/MTU. Therefore, it may be reasonable to use 1% as the strain limit for 70 GWD/MTU SNF. Furthermore, as we state in the closing paragraph of FSAR Subsection 4.A.3, burnup levels above 63.5 GWD/MTU, consistent with the behavior of all irradiated materials, would be expected to reduce zircaloy's "rate of creep", φ, for a given state of stress and temperature. In other words, a zircaloy cladding at 70 GWD/MTU burnup should reasonably be expected to exhibit a somewhat smaller rate of creep than SNF at 63.5 GWD/MTU burnup. Therefore, the total accumulated creep in 40 years, ε_s, should be less for the 70 GWD/MTU burnup fuel than its 63.5 GWD/MTU counterpart. Furthermore, it should be recalled that, by assuming 40 years, which is twice the 20-year license limit, we have imputed an additional margin of safety in the computed value of accumulated creep.

In summary, the accumulated creep, ε_s, for a 70 MWD/MTU burnup SNF is expected to be much less than 1% in the 20-year license period, if the permissible cladding temperature (PCT) is computed using a 40-year storage life and the creep rate φ is based on a correlation that is conservatively constructed using the available high burnup SNF data at this time. Therefore, the PCT limits developed for 63.5 GWD/MTU SNF can be conservatively applied to 70 GWD/MTU SNF as well.

[1] Fuketa, et al., "Behavior of PWR and BWR Fuels During Reactivity Initiated Accident Conditions", LWR Fuel Performance Meeting, Park City, Utah, April 10-13, 2000.

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Response #2

Our response to RAI 4.A.6 was premised on the following technical facts that we summarize below with supporting references, as necessary.

i. Irradiation of zircaloy cladding in the nuclear reactor increases its tensile strength. Hydride lenses formed on the cladding render the cladding non-homogeneous (i.e., the elastic properties may vary from point to point). The strength of the fuel cladding to withstand internal pressure, despite the non-homogeneity introduced by the hydride lenses in the cladding, is essentially undiminished.[2]

ii. The state of stress in the cladding that governs the rate of creep is the hoop stress. The magnitude of hoop stress is defined by equilibrium (Lame's formula), and will be the same whether or not hydride lenses are present. In other words, the state of hoop stress in the cladding will be circumferentially constant, whether or not the cladding material has localized hydrides. Hydrided regions will resist the applied hoop stress in the same manner as an unirradiated fuel cladding would.

The ability of high burnup hydrided zircaloy cladding to withstand considerable strains is supported by the test results of Goll et. al. (FSAR Appendix 4.A Reference [4.A.10]). In this work, up to 64,000 MWD/MTU burned fuel cladding was tested at very high stresses (400 MPa to 600 MPa) without rupture below 2% strain. The authors concluded that, "... the increased hydrogen content has no adverse effect on the cladding performance even at long storage times." In fact there is evidence that the influence of hydrides on the zircaloy cladding creep resistance characteristics may be beneficial: The French work by Bouffioux and Rupa[2] shows that Hydrogen (100 ppm to 1000 ppm) increases the creep resistance of Ziraloy-4 by as much as a factor of four.

Response #3

Off-normal and accident conditions germane to the integrity of the fuel cladding can be divided into two categories:

i. Those that produce additional cladding stress due to a short-duration mechanical loading.

ii. Those that produce thermal cycling of the cladding material with potential of altering its microstructure.

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[2] P. Bouffioux and N. Rupa, "Impact of Hydrogen on Plasticity and Creep of Irradiated Zircaloy-4 Cladding Tubes", 12th International Symposium, STP 1354 (1998), pages 399-420.

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The impact from a non-mechanistic cask tip-over and postulated handling accidents are examples of mechanical loading events that subject the fuel cladding to transient stress levels that may exceed the steady state (normal condition) value. The ability of the cladding to withstand a greater stress level is determined by its ultimate strength, which rises monotonically with burnup as stated in proposed FSAR Appendix 4.A. Therefore, the accident evaluations carried out for moderate burnup SNF and documented in the HI-STORM FSAR are also applicable to high burnup fuel with even greater safety margins.

The thermal event that is of potential concern to the long-term integrity of the SNF is the vacuum drying operation that results in elevated cladding temperature, albeit for a short duration. Short-term exposure to high temperatures (say, approximately 500°C), hydride re-orientation (from tangential to radial) can occur[3], which is undesirable. To protect high burnup SNF against hydride re-orientation, the proposed CoC calls for moisture removal from the MPC through a closed loop forced helium flow process.

Other short-term events such as postulated fire, blocked duct, etc., produce a short-term increase in the cladding hoop stress, which a high burnup fuel, because of its generally increased elastic strength properties, is at least as well, if not better equipped, to withstand than its low burnup counterpart. Therefore, failure of the fuel cladding leading to dispersion of its contents within the MPC can be categorically ruled out. The MPC has been structurally engineered, as documented in the HI-STORM FSAR, to provide confinement of the contained radioactive material under all conditions of storage postulated in the licensing and design basis.

Response #4

The proposed FSAR changes will be provided in Proposed FSAR Revision 1C, consistent with the RAI response.

Response #5

The verbiage in question will be removed.

[3] Kese, Kwadwe, "Hydride Re-Orientation in Zircaloy and its Effect on the Tensile Properties", SKI Report 98:32, August 1998.

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ADDITIONAL ISSUES

Response #1

The material on Pre-operational Testing and Training Exercises will be returned to the CoC as requested (Supplement 1 submittal).

Response #2

The editorial improvement will be made (Supplement 1 submittal).