HOLTEC DECOMMISSIONING INTERNATIONAL

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December 21, 2022

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Ekaterina Lenning, Project Manager
Division of Fuel Management
Office of Nuclear Material Safety and Safeguards

Subject: Submittal of Topical Report HI-2200750 "-A" Version

Reference: [1] Final Safety Evaluation by The Office of Nuclear Reactor Regulation for the Topical Report HI-2200750, Revision 0, "Holtec Spent Fuel Pool Heat Up Calculation Methodology" Docket: 99902086 (EPID: L-2020-TOP-0056) (ML22075A308)

Ms. Lenning,

Holtec greatly appreciates the staff's review and approval of the Topical Report HI-2200750 related to the Holtec Spent Fuel Pool Heat Up Calculation Methodology in Reference 1. In accordance with the request with that approval, this letter provides the proprietary and nonproprietary topical report "-A" version. Also included is an affidavit requesting that the proprietary topical report be withheld from public disclosure.

Should you have any questions or require further information, please contact William Noval, Holtec Decommissioning International Director of Regulatory Affairs at (856) 797-0900 x3587.

Sincerely,

Jean A. Fleming
Digitally signed by Jean A. Fleming
Date: 2022.12.21 13:47:07 -05'00'

Jean A. Fleming
Vice President, Licensing, Regulatory Affairs, & PSA
Holtec International

Attachments:
Attachment 1: HI-2200750-A Proprietary Version
Attachment 2: HI-2200750-A Nonproprietary Version
Attachment 3: Affidavit Pursuant to 10 CFR 2.390

ATTACHMENT 1

HI-2200750-A
Proprietary Version
Withhold from public disclosure under 10 CFR 2.390

TOPICAL REPORT FOR SPENT FUEL POOL HEAT UP CALCULATION METHODOLOGY

By

Holtec International
Holtec Technology Campus
One Holtec Boulevard
Camden, NJ 08104
(holtecinternational.com)

Docket No. 99902086
Holtec Project 3131
Holtec Report No. HI-2200750

ATTACHMENT 2

HI-2200750-A
Non-Proprietary Version

TOPICAL REPORT
FOR SPENT FUEL POOL HEAT UP CALCULATION METHODOLOGY

By

Holtec International
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One Holtec Boulevard
Camden, NJ 08104
(holtecinternational.com)

Docket No. 99902086
Holtec Project 3131
Holtec Report No. HI-2200750

TABLE OF CONTENTS

SECTION A

HOLTEC INTERNATIONAL

Holtec Technology Campus, One Holtec Blvd, Camden, NJ 08104
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September 29, 2020

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 99902086 - HDI Spent Fuel Pool Heat Up Calculation Methodology

Subject: Submittal of Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report

Dear Sir or Madam:

Holtec is pleased to submit for NRC review and approval a topical report that provides an expanded methodology for calculating spent fuel pool thermal conditions. This methodology is straight-forward and validated against NRC-approved calculational methodologies, as well as NRC methods themselves. Due to the improvement in safety that can be realized through use of the methodology and its straightforward and robust validation, we request that the NRC staff review and approve this topical report by March 31, 2021.

The report is included as Attachment 1 this letter. Since this document is considered proprietary, Attachment 2 includes an affidavit according to 10CFR2.390 requesting that it be withheld from public disclosure.

If you have any questions, please contact me at 856-797-0900 ext. 3813.

Sincerely,

Andrea L. Sterdis

Digitally signed by Andrea L. Sterdis
DN: cn=Andrea L. Sterdis, c=US, o=HDI, ou=Holtec Decommissioning International, email=andrea.sterdis@holtec.com
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Andrea L. Sterdis
VP, Regulatory and Environmental Affairs
Holtec Decommissioning International

Holtec Technology Campus, One Holtec Blvd, Camden, NJ 08104
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Attachments:

Attachment 1	Method for Determining Spent Fuel Assembly Heat Up During a Theoretical Drain Down Event, HI-2200750, Revision 0 (Holtec Proprietary)
Attachment 2:	Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure

cc:

Robert Lucas, NRC, NRR/DORL/LLPB
Dennis Morey, NRC, NRR/DORL/LLPB
Ekaterina Lenning, NRC, NRR/DORL/LLPB
Christopher Regan, NRC, NMSS/DFM

Revision Log

Revision	Description of Changes
0	Initial issue.

EXECUTIVE SUMMARY

Holtec International has developed a conservative method for calculating the transient heat up of spent fuel assemblies under a hypothetical spent fuel pool drain down event. Following defueling of the reactor, the ignition of the zirconium cladding of the spent fuel assembly within the spent fuel pool due to a loss of spent fuel pool cooling becomes one of the highest consequence potential accidents. Traditional NRC approved adiabatic calculations, such as the ones presented in NUREG-1738 [1], [

] 4.a, 4.b will take longer than 10 hours to reach a temperature associated with a zirconium fire (zirc fire). These calculations are very conservative, [

] 4.a, 4.b In other words, [

] 4.a, 4.b This approach can be modified to [

] 4.a, 4.b The results from this modified approach can [

] 4.a, 4.b By using this method [ ] 4.a, 4.b a significant and real safety benefit can be recognized.

A recent NRC study RES/DSA/FSCB 2016-03 [2] explores the conservatisms embodied by the adiabatic calculation and summarizes the results of several strategies for calculating a more realistic heat up time. Principal among these are inclusion of the mass of the spent fuel racks in the heat up equation and, through use of the MELCOR code, introduction of convective heat dissipation. [

] 4.a, 4.b following a post shut-down drain down event in the pool. [

] 4.a, 4.b

The results presented in this report are shown to be consistent with those found in RES/DSA/FSCB 2016-03 [2]. Holtec further validated this method by performing the Computational Fluid Dynamics (CFD) calculations presented in Appendix D. [

] 4.a, 4.b

These CFD calculations further demonstrate the conservative nature of the method presented in this report, as the CFD results illustrate the significant levels of conservative margin in the proposed method.

Based upon the results of the method presented in this report, as validated against the NRC and Holtec CFD calculations, [ ] [4.a, 4.b] and can be leveraged to [ ] [4.a, 4.b]

Table of Contents

List of Tables

List of Figures

1.0 PURPOSE

Spent fuel pools safely store spent nuclear fuel (SNF) from nuclear power plants in fuel racks. The fuel racks are cellular structures with design features that facilitate the circulation of water through the storage cells to maintain the fuel rods in a cooled state. The racks are staged on the pool's slab with an elevated baseplate which supports the fuel assemblies and provides a bottom plenum of water that feeds the coolant to each storage cell. High reactivity fuel or fuel recently removed from the reactor is stored in flux-trap (FT) type fuel racks. Flux-trap type racks are characterized by a small lateral gap around each storage cell. Other fuel assemblies can be stored in the so-called non-flux trap (NFT) type racks which feature no gap between adjacent storage cells. Most PWR plants have racks of both flux trap and non-flux trap types[1]. BWR fuel, due to its relatively lower reactivity, is typically stored in NFT racks only.

Because keeping the fuel cool is a principal function of fuel racks, they are equipped with appropriate features to promote efficient fuel decay heat removal. The bulk temperature of the water in the pool is maintained in a desired range using a set of pumps and heat exchangers with redundancy to ensure reliable cooling. The pool's heat removal system and robust design that protect against abnormal conditions have proven to be reliable, making in-pool storage a credibly safe means to store the fuel.

Once a reactor is permanently defueled, a drain down of the water in the spent fuel pool, while highly unlikely, has one of the highest consequences of the remaining potential accidents in terms of public health and safety. Because the heat rejection rate via air cooling is several orders of magnitude smaller than via water cooling, the fuel temperature will rise until the heat generation rate equilibrates with the heat dissipation rate. It is known that the zirconium fuel cladding becomes sensitive to combustion in an oxygen environment at temperatures above 900°C. To reduce or prevent the fuel's vulnerability to a zirconium fire (zirc fire) during a drain down event of the pool, it is desirable to adopt measures to minimize the fuel cladding's temperature rise in the air environment. In the event that preventing exceedance of the allowable limit is not possible, as is the case immediately following reactor shutdown, it is essential [

] 4.a, 4.b

Historically, permanently shut down plants have computed the period of zirc fire vulnerability after a reactor's shutdown, to establish the basis for adjusting requirements that are needed to address the hypothetical zirc fire event. Such analyses have been often performed [

] 4.a, 4.b

[1] The racks with FT are commonly referred to as Region 1 racks and the NFT types are referred to as Region 2 racks. This terminology is adopted for the remainder of the report.

The objective of this topical report is to present a generic methodology that [

] 4.a, 4.b This method can thus be used to [

] 4.a, 4.b

It is shown in this report that:

i. [

] 4.a, 4.b

ii. The method developed maintains large levels of conservatism through its assumptions.

iii. The method can be used to analyze any fuel pool with any kind of fuel.

iv. The method's results are consistent with the findings of more complex and sophisticated analyses performed both by the NRC using a MELCOR model and by Holtec using a Computational Fluid Dynamics (CFD) model.

[

] 4.a, 4.b

2.0 ACCEPTANCE CRITERIA

While the method presented in this report does not itself have a specific acceptance criteria, it is intended for use in determining the elapsed time until loss of spent fuel pool water cooling results in the ignition of the spent fuel assembly zirconium cladding. NUREG-1738 [1] establishes 900ºC as a reasonable temperature limit to protect against a runaway oxidation, and thus a zirc fire, in an air environment, in combination with a 10-hour time window to take actions. SECY-99-168 [3] deems an elapsed time of at least ten hours as being generally sufficient time from the loss of spent pool cooling "to take mitigative actions and, if necessary, offsite protective measures without preplanning" to the point that the risk and consequence of a zirc fire is sufficiently diminished to warrant adjustment of a nuclear plant's emergency plan.

As such, the results presented in Section 7.0 of this report [

] 4.a, 4.b to reach 900ºC is ten hours or longer.

3.0 METHODOLOGY

3.1 Overview

This section presents the methodology used to calculate the spent fuel assembly bulk temperatures as a function of time following a theoretical drain down event. The assumptions inherent to the methodology are discussed in more detail in Section 4.0.

The methodology presented in this report considers [

] 4.a, 4.b The calculations are conservative [

[ ] 4.a, 4.b

[ ] 4.a, 4.b

] 4.a, 4 b is discussed further in Sections 3.6 and 3.7.

[ ] 4.a, 4.b

Figure 3.1: [

] 4.a, 4.b

3.2 [ ] 4.a, 4.b

[ ] 4 a, 4 b

4.a, 4.b

[ ] Eq. 3-1

Where:

[ ] 4.a, 4.b

[ ] 4.a, 4.b

4.a, 4.b

[ ] Eq. 3-2

Where:

[ ] 4.a, 4.b

[ ] 4.a, 4.b

4.a, 4.b

[ ] Eq. 3-3

[ ] 4.a, 4.b

4.a, 4.b

[ ] Eq. 3-4

Where:

[ ] 4.a, 4.b

4.a, 4.b

[ ] Eq. 3-5

Where:

[ ] 4.a, 4.b

3.3 **[ ] 4.a, 4.b**

[ ] 4.a, 4.b

[ ] 4.a, 4.b Eq. 3-6

Where:

[ ] 4.a, 4.b

4.a, 4.b

[ ] Eq. 3-7

Where:

4.a, 4.b

Eq. 3-8

4.a, 4.b

Eq. 3-9

Where:

4.a, 4.b

Eq. 3-10

[ ] 4.a, 4.b

Eq. 3-11

[ ] 4.a, 4.b

Eq. 3-12

Eq. 3-13

[ ] 4.a, 4.b

.

Figure 3.2: [ ] 4.a, 4.b

3.4 Material Specific Heats

The specific heats of the UO_2 and zircaloy are determined based on the methods presented in NUREG/CR-0479. The specific heats of UO_2 are calculated from the specific heat capacity correlation provided in Appendix A of NUREG/CR-0479 [5] at each time step for each fuel assembly. This correlation is reproduced below:

$$C_{p_{UO_2}} = \frac{K_1 \theta^2 e^{\theta/T}}{T^2\left(e^{\theta/T} - 1\right)^2} + K_2 T + \left(\frac{O/M}{2}\right)\frac{K_3 E_D}{RT^2} e^{-E_D/RT} \qquad \text{Eq. 3-14}$$

Where:

K_1, K_2, K_3, E_D, and θ are each constants.
T is temperature in K
O/M is the oxygen to metal ratio of UO_2
R is the gas constant.

The specific heats of zircaloy are linearly interpolated based upon the experimental data presented in Appendix B of NUREG/CR-0479. [5] These results are tabulated in intervals of 5ºC in Appendix A of this report. For each timestep in the calculation, the specific heat of zircaloy at the nearest lower bound temperature increment are used to determine the incremental temperature increase. This approach yields conservative results as the specific heat of zircaloy in this temperature range increase with rising temperature.

3.5 Assembly Material Weights

In instances where the masses of UO_2 or zircaloy within an assembly are not directly available, these values can be calculated from other assembly characteristics. The mass of UO_2 in the fuel assembly is estimated by scaling the mass of heavy metal by the mass fraction of U in UO_2, which simply multiplies the mass of heavy metal by a factor of 1.134.

The mass of zircaloy in the assembly is calculated as:

$$m_z = \rho_z L_{AR} N_r \frac{\pi}{4}(OD^2 - ID^2) \qquad \text{Eq. 3-15}$$

Where:

ρ_z is the density of zirconium
L_{AR} is the length of the active region
N_r is the typical number of rods per assembly
π is the mathematical constant
OD is the outer diameter of the fuel rod cladding.
ID is the inner diameter of the fuel rod cladding.

It is noted that these methods for determining material weights were utilized in this report based on the materials available to be referenced, and alternative approaches for determining appropriate inputs are acceptable.

3.6 [ ] 4.a, 4.b

[ ] 4.a, 4.b

[ ] 4.a, 4.b

3.7 [ ] 4.a, 4.b

[4.a, 4.b

Eq. 3-16]

4.a, 4.b

4.a, 4.b

Figure 3.3:

4.a, 4.b

4.0 ASSUMPTIONS AND CONSERVATISMS

4.1 Conservatisms

4.1.1 Only the masses of the UO_2 fuel pellets and the zircaloy cladding are considered for heat up in this method. This approach conservatively neglects the mass of the spent fuel racks and other fuel assembly components, which in reality would also absorb the generated heat, further slowing the rate of increase in temperature of the spent fuel assemblies. Consideration of the racks alone has been shown in RES/DSA/FSCB 2016-03 [2] to further extend heat up times by 35-40% in BWRs and 15-20% in PWRs.

4.1.2 [ ] 4.a, 4.b

] 4.a, 4.b consistent with those used in NUREG-1738 [1], RES/DSA/FSCB 2016-03 [2], and numerous others.

[ ] 4.a, 4.b

4.1.3 [ ] 4.a, 4.b

4.1.4 [ ] 4.a, 4.b

4.1.5 The starting time of the calculation is assumed to occur when the spent fuel pool is drained. This conservatively neglects the significant period of time where a leak or other loss of spent fuel cooling would have been detected and plant staff would be able to begin taking mitigating action. Inherent in this approach is the presumption that the spent fuel pool water instantaneously vaporizes, whereas in reality the mass of spent fuel pool water is still present to absorb and dissipate heat from the spent fuel assemblies. This conservatism is consistent with that assumed in the calculations described in SECY-99-168 [3].

4.2 Assumptions

4.2.1 [

] 4.a, 4.b This is consistent with the approach taken for adiabatic calculations in NUREG-1738 [1], RES/DSA/FSCB 2016-03 [2], and other prior licensee calculations. It [ ] 4.a, 4.b

4.2.2 The spent fuel assembly [

] 4.a, 4.b This assumption is reasonable as the fuel and cladding are in close contact, and [

] 4.a, 4.b This is also a common assumption for calculations of this nature and is consistent with NUREG-1738 [1] and RES/DSA/FSCB 2016-03 [2]. In BWR fuel, the fuel channel is similarly assumed [ ] 4.a, 4.b

4.2.3 [ ] 4.a, 4.b

4.2.4 [ ] 4.a, 4.b

4.2.5 The starting temperatures of the spent fuel assemblies [

] 4.a, 4.b This choice of starting temperature is realistic and appropriate, [

] 4.a, 4.b

5.0 INPUT DATA

5.1 Generic Input Data

The subsequent tables present the standardized coefficients and values associated with the equations provided in Section 3.0.

Table 5.1: Constants Used in UO_2 Specific Heat Capacity Correlations

Parameter	Value [Ref]
K_1	296.7 J/kg/K [5]
K_2	2.43 x 10^{-2} J/kg/K^2 [5]
K_3	8.745 x 10^7 J/kg [5]
Θ	535.285 K [5]
E_D	1.577 x 10^5 J/mol [5]
O/M of UO_2	2.0 [8]
R	8.3143 J/mol/K [5]

Table 5.2: Additional Coefficients and Parameters

Parameter	Value [Ref]
[ ] 4.a, 4.b	[ ] 4.a, 4.b
[ ] 4.a, 4.b	[ ] 4 a, 4.b
[ ] 4.a, 4.b	[ ] 4.a, 4 b
Density of Zircaloy, ρ_z	6490 kg/m^3 [5]

5.2 Site Specific Input Data

The subsequent tables present the site-specific values used to generate the example results in Section 7.0.

Table 5.3: Example Site Fuel Assembly Parameters

Parameter	Example Plant A (W 15x15 OFA)	Example Plant B (CE 15x15 Palis. PWR)	Example Plant C (GE GNF2)
Active Length	3.657 m [8]	3.098 m [10]	3.689 m [11]
[ ] 4.a, 4.b	[ ] 4 a, 4.b	[ ] 4.a, 4.b	[ ] 4.a, 4 b
Typical Number of Rods	204 [8]	204 [8]	---
[ ] 4.a, 4.b	[ ] 4.a, 4.b	[ ] 4.a, 4.b	[ ] 4.a, 4.b
Cladding Outer Diameter	10.718 mm [12]	10.591 mm [10]	10.261 mm [11]
Cladding Inner Diameter	9.474 mm [12]	9.321 mm [10]	---
[ ] 4.a, 4.b	[ ] 4.a, 4.b	[ ] 4.a, 4 b	[ ] 4.a, 4 b
[ ] 4.a, 4.b	[ ] 4 a, 4.b	[ ] 4.a, 4.b	[ ] 4.a, 4.b
Mass of Heavy Metal	462.664 kg [8]	412.769 kg [8]	---
Mass of UO_2	524.660 kg *	468.080 kg *	208.011 kg [11]
Mass of Zircaloy	95.516 kg *	81.461 kg *	41.666 kg [11]

Notes:
1 – Some values in the table are converted from the values in the provided reference to the units stated here, rounded as shown.
2 – For some parameters, multiple values can be found in the listed reference. In that case, the most conservative (smallest) value is used.
3 – Values marked by an asterisk (*) are calculated from other values provided in this table and Table 5.2.

[

] 4.a, 4.b

6.0 EXAMPLE CALCULATIONS

Example calculations for two representative cases are presented in Appendix B and Appendix C
.

7.0 RESULTS OF EXAMPLE CALCULATIONS

7.1 [ ] 4.a, 4.b

[ ] 4.a, 4.b

Figure 7.1: [ ] 4.a, 4.b

[ ] 4.a, 4.b

4.a, 4.b

[ ]

Figure 7.2: [

]4.a, 4.b

7.2 Interpretation of Results

Based upon the results in Section 7.1, [

] 4.a, 4.b While the proposed method could be used to [

] 4.a, 4.b which in turn introduces additional conservatism.

As described in Sections 3.6 and 3.7, [ ] 4.a, 4.b

7.3 [ ] 4.a, 4.b

[ ] 4.a, 4.b

7.4 Results for Example Plants

Table 7.1 presents the results for three example plants. [ ] 4.a, 4.b

Table 7.1: Calculated Heat Load Limits for Example Plants

[ ] 4.a, 4.b

4.a, 4.b

7.5 Comparison to NRC Calculations

4.a, 4.b

]4.a, 4.b The MELCOR calculations were performed using the average power from Figure 1 in [2] for a BWR assembly with a burnup of 60 GWd/MTHM across a variety of cooling times. [

]4.a, 4.b

In order to closely replicate the scenarios evaluated, the heat load values used in the study were estimated from the respective figure, [

]4.a, 4.b Calculations were performed using these same heat load values used by [2] under the developed method for Example Plant C, also a BWR. These results are presented beside the MELCOR results in Figure 7.3. [

]4.a, 4.b This trend is indicative of significant conservatism [

]4.a, 4.b This further illustrates the conservatism of the method proposed in this report as a whole.

Table 7.2: [**]4.a, 4.b**

4.a, 4.b

[ ] 4.a, 4.b

[ ] 4.a, 4.b

Figure 7.3: [ ]4.a, 4.b

7.6 Comparison to Computational Fluid Dynamics (CFD) Calculations

As discussed in Appendix D , CFD calculations were performed to further confirm the conservative nature of the model, [ ]4.a, 4.b As shown by the comparison of the CFD cases with the corresponding results from Figure 7.2 and Table 7.1, the method presented in this report results in shorter heat up times and is therefore conservative.

8.0 CONCLUSION

This topical report presents a simple and effective method for determining PWR and BWR spent fuel assembly heat up in spent fuel pools during a hypothetical drain down event. Results are

compared against more sophisticated models and show significant levels of conservatism, [

][4.a, 4.b]

9.0 REFERENCES

[1] "NUREG-1738 Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (ML010430066)," 2001.

[2] "RES/DSA/FSCB 2016-03 Spent Fuel Assembly Heat Up Calculations in Support of Task 2 of User Need NSIR-2015-001 (ML16110A431)," 2016.

[3] "SECY-99-168 Improving Decommissioning Regulations for Nuclear Power Plants (ML992800087)," 1999.

[4] S. Middleman, An Introduction to Mass and Heat Transfer, New York: John Wiley & Sons, Inc., 1998.

[5] "NUREG/CR-0479 MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," 1981.

[6] "NSIR/DPR-ISG-02 Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants (ML14106A057)," 2015.

[7] EPRI, "Severe Accident Management Guidance Technical Basis Report, Volume 2: The Physics of Accident Progression," 2012.

[8] "DOE/RW-0184-Vol. 3 Characteristics of Spent Fuel, High-Level Waste, and other Radioactive Wastes Which May Require Long-Term Isolation," 1987.

[9] Holtec International, "HI-2002444R20 Final Safety Analysis Report for the HI-STORM 100 Cask System," 2020.

[10] "Palisades Final Safety Analysis Report, Revision 29, Table 3-2".

[11] "Response to Request for Additional Information (RAI) and Supplemental Information Regarding Request for Changing Emergency Preparedness License Amendment No. 294 Effective Date (Change to Adiabatic Heat-up Calculation) (ML19044A643)," 2019.

[12] "EN-DC-141R17-190605, Design Input Record, Attachment-9.1".

[13] "C-1302-226-E310-459R0 Oyster Creek Reactor Building Heat Up and Fuel Cladding Temperature for Drained Spent Fuel Pool," 2019.

[14] "Regulatory Improvements for Power Reactors Transitioning to Decommissioning (ML15026A316)," 2017.

[15] "NUREG-2161 Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor (ML14255A365)," 2014.

Appendix A TABULATION OF ZIRCALOY SPECIFIC HEAT VALUES BY TEMPERATURE

The specific heats of zircaloy are linearly interpolated based upon the experimental data presented in Appendix B of NUREG/CR-0479. [5]

K	C	J/kg/K
298	25	280.6
303	30	281.6
308	35	282.7
313	40	283.7
318	45	284.8
323	50	285.8
328	55	286.9
333	60	287.9
338	65	289.0
343	70	290.0
348	75	291.1
353	80	292.1
358	85	293.2
363	90	294.2
368	95	295.3
373	100	296.3
378	105	297.4
383	110	298.4
388	115	299.5
393	120	300.5
398	125	301.6
403	130	302.4
408	135	303.0
413	140	303.6
418	145	304.2
423	150	304.8
428	155	305.4
433	160	306.0
438	165	306.6
443	170	307.2
448	175	307.8
453	180	308.4
458	185	309.0
463	190	309.6
468	195	310.2
473	200	310.8
478	205	311.4
483	210	312.0
488	215	312.6
493	220	313.2
498	225	313.8
503	230	314.4
508	235	315.1
513	240	315.7
518	245	316.3
523	250	316.9
528	255	317.5
533	260	318.1
538	265	318.7
543	270	319.3
548	275	319.9
553	280	320.5
558	285	321.1
563	290	321.7
568	295	322.3
573	300	322.9
578	305	323.5
583	310	324.1
588	315	324.7
593	320	325.3
598	325	325.9
603	330	326.5
608	335	327.1

613	340	327.7
618	345	328.3
623	350	328.9
628	355	329.6
633	360	330.2
638	365	330.8
643	370	331.3
648	375	331.8
653	380	332.3
658	385	332.8
663	390	333.2
668	395	333.7
673	400	334.2
678	405	334.7
683	410	335.2
688	415	335.7
693	420	336.2
698	425	336.7
703	430	337.2
708	435	337.6
713	440	338.1
718	445	338.6
723	450	339.1
728	455	339.6
733	460	340.1
738	465	340.6
743	470	341.1
748	475	341.6
753	480	342.0
758	485	342.5
763	490	343.0
768	495	343.5
773	500	344.0
778	505	344.5
783	510	345.0
788	515	345.5
793	520	346.0
798	525	346.4

803	530	346.9
808	535	347.4
813	540	347.9
818	545	348.4
823	550	348.9
828	555	349.4
833	560	349.9
838	565	350.4
843	570	350.8
848	575	351.3
853	580	351.8
858	585	352.3
863	590	352.8
868	595	353.3
873	600	353.8
878	605	354.3
883	610	354.8
888	615	355.2
893	620	355.7
898	625	356.2
903	630	356.7
908	635	357.2
913	640	357.7
918	645	358.2
923	650	358.7
928	655	359.2
933	660	359.6
938	665	360.1
943	670	360.6
948	675	361.1
953	680	361.6
958	685	362.1
963	690	362.6
968	695	363.1
973	700	363.6
978	705	364.0
983	710	364.5
988	715	365.0

993	720	365.5
998	725	366.0
1003	730	366.5
1008	735	367.0
1013	740	367.5
1018	745	368.0
1023	750	368.4
1028	755	368.9
1033	760	369.4
1038	765	369.9
1043	770	370.4
1048	775	370.9
1053	780	371.4
1058	785	371.9
1063	790	372.4
1068	795	372.8
1073	800	373.3
1078	805	373.8
1083	810	374.3
1088	815	374.8
1093	820	502.0
1098	825	524.0
1103	830	546.0
1108	835	568.0
1113	840	590.0
1118	845	596.3
1123	850	602.5
1128	855	608.8
1133	860	615.0
1138	865	641.0
1143	870	667.0
1148	875	693.0
1153	880	719.0
1158	885	743.3
1163	890	767.5
1168	895	791.8
1173	900	816.0

Appendix B EXAMPLE CALCULATION [ ][4a., 4.b] AT EXAMPLE PLANT A

4.a, 4.b

[

]

[ ]

4.a, 4.b

4.a, 4.b

Appendix C EXAMPLE CALCULATION [ ][4a., 4.b] AT EXAMPLE PLANT A

4.a, 4.b

[ ]

4.a, 4.b

[

]

4.a, 4.b

Appendix D Validation Studies Using CFD Method

A conservative method for calculating the transient heat up of spent fuel assemblies under a hypothetical spent fuel pool drain down event is presented in the main part of the report. This appendix presents an independently conducted analysis, utilizing a different method than in the main report, which confirms the conservative nature of that method in determining fuel pool heat up rates. The independent validation method presented in the appendix is a widely used Computational Fluid Dynamics (CFD) approach. This approach has a long history of use in solving a wide array of thermal-hydraulic problems in wet and dry storage of spent nuclear fuel (SNF). The CFD code, FLUENT, used since the mid-1990s in virtually every Holtec docket, [

][4.a, 4.b] The concept is further discussed in the next sections of this appendix. [

][4.a, 4.b] presented in main part of the report, are analyzed in Section D.4 and the results from the two methods are compared.

D.1 EVALUATION METHODOLOGY

D.1.1 Overview

The CFD approach has a long history of use in solving a wide array of thermal-hydraulic problems in the nuclear industry. The computer code ANSYS FLUENT is a Holtec QA-validated code [D.6.4] used for this purpose. CFD owes its origin to the safety analyses performed for Connecticut Yankee [D.6.1], Millstone Unit 1 [D.6.2] and Vermont Yankee [D.6.3] which utilized the same code (ANSYS FLUENT [D.6.4], [D.6.5]) using simplified models. Additionally, the thermal modeling methodology described in this section has been benchmarked against test data from dry storage casks containing SNF as described in Section D.3 of this appendix. The benchmarking evaluations show that the FLUENT solutions are conservative in all cases.

[

][4.a, 4.b] The principal features of the thermal models are described in this section and example calculations are reported Section D.4 of this appendix.

D.1.2 Methodology and Principal Assumptions

Fuel rack is a 3-D array of square shaped cells that can hold nuclear fuel assemblies. [

][4.a, 4.b] Details of the methodology are provided in the following sub-sections.

D.1.2.1 Details of the [ ]4.a, 4.b

[ ]

4.a, 4.b

4.a, 4.b

D.1.2.2 [ ][4.a, 4.b]

[ ][4.a, 4.b]

][4.a, 4.b] This approach is used in the various dockets approved by NRC (Table D.1.1).

[

][4.a, 4 b] This approach is used in the various dockets approved by NRC (Table D.1.1).

[

][4.a, 4.b] This approach is used in the various dockets approved by NRC (Table D.1.1).

D.1.2.3 [ ][4.a, 4.b]

[ ]

[ ] 4.a, 4.b

[ ] 4.a, 4.b Eq (D.1.1)

where:

[ ] 4.a, 4.b

D.1.2.4 [ ] 4.a, 4.b

[ ] 4.a, 4.b

[ ] 4.a, 4.b Eq (D.1.2)

[ ] 4.a, 4.b

4.a, 4.b

Eq (D.1.3)

where,

4.a, 4.b

4.a, 4.b

Eq (D.1.4)

where,

4.a, 4.b

4.a, 4.b

Eq (D.1.5)

where,

[ ] 4.a, 4.b

[ ] 4.a, 4.b

Eq (D.1.6)

Where,

[ ] 4.a, 4.b

[ ] 4.a, 4.b

Eq (D.1.7)

Where,

[ ] 4.a, 4.b

D.1.2.5 Inputs

The following inputs are adopted for CFD analyses [

]4.a, 4.b using the models and methodology described in the above sub-sections:

[

] 4.a, 4.b

* [

]4.a, 4 b

Table D.1.1: List of Prior NRC Approval on CFD Methodology

Sl. No	Description	NRC Docket No.	Location in Approved Docket
1.	[ ]4.a, 4 b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
		71-9325	Section [ ]4 a, 4.b
		71-9367	Section [ ]4 a, 4.b
		71-9373	Section [ ]4 a, 4.b
		71-9374	Section [ ]4 a, 4.b
2.	[ ]4.a, 4.b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
3.	[ ]4.a, 4.b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
4.	[ ]4.a, 4.b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
5.	[ ]4.a, 4 b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
6.	[ ]4.a, 4 b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
		71-9325	Section [ ]4 a, 4.b
7.	[ ]4.a, 4.b	72-1014	Section [ ]4 a, 4.b
		72-1032	Section [ ]4 a, 4.b
		72-1040	Section [ ]4 a, 4.b
		71-9325	Section [ ]4 a, 4.b
		71-9367	Section [ ]4 a, 4.b
		71-9373	Section [ ]4 a, 4.b

Table D.1.2: Reference Fuel Parameters [

]4.a, 4.b

[ ] 4.a, 4.b

Notes:

1. **[**

]4.a, 4.b

Table D.1.3: Normalized Distribution Based on Burnup Profile [D.6.12]

PWR DISTRIBUTION		
Interval	Axial Distance from Bottom of Active Fuel (% of Active Fuel Length)	Normalized Distribution
1	0% to 16-2/3%	0.88755
2	16-2/3% to 33-1/3%	1.1050
3	33-1/3% to 50%	1.0980
4	50% to 66-2/3%	1.0790
5	66-2/3% to 83-1/3%	1.0501
6	83-1/3% to 100%	0.7804

[ ]4.a, 4.b

Figure D.1.1: [ ]4.a, 4.b

4.a, 4.b

[

]

Figure D.1.2: [ ]4.a, 4 b

4.a, 4.b

Figure D.1.3(a): [**]** 4.a, 4.b

[ ]4.a, 4.b

Figure D.1.3(b): [ ]4.a, 4.b

D.2 ACCEPTANCE CRITERIA

The acceptance criterion stated in NUREG-1738 [D.6.6] is adopted herein. The bulk average temperature of active fuel region of a fuel assembly shall not exceed 900^{o}C after 10 hours of dry cooling.

D.3 BENCHMARKING AND VALIDATION OF CFD METHODOLOGY

The Computational Fluid Dynamics (CFD) approach has a long history of solving a wide range of thermal analysis problems in the nuclear industry. The computer code ANSYS FLUENT is a QA-validated code [D.6.5] used for this purpose. Additionally, this code has been validated under Holtec's QA program [D.6.10]. Numerous wet and dry storage thermal-hydraulic evaluations have been performed using FLUENT and have been approved by the NRC (Table D.4.1).

The FLUENT thermal modeling methodology has been benchmarked using data from tests conducted with dry storage casks loaded with SNF. The benchmark work is archived in QA-validated Holtec reports[2]. A summary of a few benchmarking studies is presented below.

D.3.1 Benchmark against Test Data from TN-24P Cask

The FLUENT thermal modeling methodology has been benchmarked with full-scale cask test data (EPRI TN-24P cask testing), as well as with PNNL's COBRA-SFS modeling of the HI-STORM System [D.6.9]. The 3D thermal modeling methodology adopted in the benchmarking analysis is similar, as described below, to that used in the CFD evaluations presented in this Appendix.

1. [ ]4.a, 4.b
2. [ ]4.a, 4.b
3. [ ]4.a, 4.b
4. [ ]4.a, 4.b

The measured cladding temperature from the test data are compared against predicted temperatures from the simulations. Test data documents results from various scenarios such as vertical orientation, pressurized gas, helium backfilled, nitrogen backfilled. All these scenarios were evaluated, and the results demonstrate that the thermal analysis methodology conservatively predicts the cladding temperatures measured. Also, the variation of the predicted cladding temperatures along the height of the fuel assemblies compared favorably with the experimental data. The benchmark work is archived in a QA-validated Holtec report [D.6.11]. This provides assurance that the CFD modelling approach is appropriate for evaluating the time it takes to achieve a temperature of 900ºC associated with a zirc fire.

D.3.2 Benchmark against Test Data from Holtec's Dry Storage System

Holtec has also performed a thermal validation test on a dry storage system loaded with SNF, as required by Holtec's dry cask storage system's Technical Specification [D.6.7]. As part of this validation, the total air mass flow rate through the cask system using direct measurements of air

[2] Specific references are cited in the following sub-sections.

velocity in the cask inlet vents was measured. An analysis of the cask system was performed using the CFD method [D.6.15], consistent with that outlined in this appendix, and results compared with the measurements. Uncertainties in the numerical and experimental studies were included in this comparison.

All analyzed scenarios presented in the report [D.6.15] conservatively bound the mass flow rate determined through the experimental methods.

D.3.3 Benchmarking Conclusions from NUREG

NRC has independently performed benchmarking studies against experimental data from casks loaded with SNF. NRC's benchmarking was primarily performed to validate the ANSYS Fluent CFD code and methodology [D.6.14] using the experimental data documented in NUREG/CR-7250 [D.6.13], to test the validity of the modeling presently used to determine cladding temperatures and air mass flow rates in vertical dry casks. Multiple data sets of varying conditions were collected and analyzed for these efforts, as described in NUREG/CR-7250.

The 3D CFD modeling methodology adopted in the NUREG studies is similar to that used in the CFD evaluations presented in this Appendix. CFD models were built for each of these configurations and tested under the same conditions to support the validation study. Steady-state and transient simulations were performed at different decay heat power values and canister helium pressures. PCT, temperature profiles for different wall structures (i.e., channel box, basket, and pressure vessel), and air mass flow rate from the CFD predictions were compared to the experimental data. The CFD results and experimental data for PCT and air mass flow rate agreed favorably within the calculated validation uncertainty for all the cases [D.6.14].

Table D.3.1: Approved Holtec Systems under NRC Dockets that use FLUENT

System Name	NRC Docket Number
HI-STORM 100 (Storage)	72-1014
HI-STORM Flood/Wind (Storage)	72-1032
HI-STORM UMAX (Storage)	72-1040
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 180D (Transportation)	71-9367
HI-STAR 60 (Transportation)	71-9336
HI-STAR 100MB (Transportation)	71-9378
HI-STAR 80 (Transportation)	71-9374
HI-STAR 190 (Transportation)	71-9373

D.4 [ ] 4.a, 4.b

To illustrate that the method presented in main part of the report is bounding, CFD calculations are performed [ ] 4.a, 4.b results from the two methods are compared. [ ] 4.a, 4.b CFD evaluations presented in this section are performed using the methodology articulated in Section D.1. The principal steps are as follows:

[ ] 4.a, 4.b

[ ] 4.a, 4.b

] 4.a, 4.b presented herein demonstrates the acceptance criteria from Section D.2 is satisfied. Additionally, all the results demonstrate that the results [ ] 4.a, 4.b in the main report are bounding.

In addition to the above, [ ] 4.a, 4.b

[ ]4.a, 4.b Therefore, these additional studies further ascertain the results presented in the main report are conservative.

Table D.4.1: [ ] 4 a, 4.b

[ ] 4.a, 4.b

Table D.4.2: Fuel Data for Plant A

Parameter	Value
Array Size	15 x 15
Fuel Cladding OD	10.72 mm
Fuel Cladding ID	9.48 mm
Fuel Pellet Diameter	9.29 mm
Fuel Rod Pitch	14.30 mm
Number of Guide Tubes	20
Number of Instrument Tubes	1
Guide Tube / Instrument Tube OD	10.72 mm
Guide Tube / Instrument Tube thickness	0.43 mm
Rod Height	3860.8 mm
Active Height	3657.6 mm
Weight of UO_2	530 kg
Weight of Zr	104 kg

Table D.4.3: Fuel Racks – Region 2 Geometrical Data for Plant A

Parameter	Value
Rack Cell ID	8.83 in
Rack Height	165 in
Bottom Rack Support Height	11 in
Flux Trap Width	Not Applicable
Neutron Absorber Thickness[3]	0.075 in

[3] [ ] 4.a, 4.b

Table D. 4.4: [ ]4.a, 4.b

4.a, 4.b

Table D.4.5: [ ]4.a, 4.b

[ ] 4.a, 4.b

Table D.4.6: [ ]4.a, 4.b

[ ] 4.a, 4.b

[ ] 4.a, 4.b

Figure D.4.1: [ ]4.a, 4.b

[ ] 4.a, 4.b

Figure D.4.2: [

] 4.a, 4.b

[ ] 4.a, 4.b

Figure D.4.3: [

] 4.a, 4.b

[ ]4.a, 4.b

Figure D.4.4: [ ]4.a, 4.b

D.5 CONCLUSIONS

The CFD predictions in all cases support the conservative nature of the fuel pool heat up calculations from the simplified model in the main body of this report. These results from the CFD model and its use of the most widely used and QA-validated code for gaseous flow conditions in the nuclear industry (FLUENT) provides the confidence that the method presented in the main report can serve as a reliable vehicle for analyzing fuel pools in a drain down condition.

D.6 REFERENCES

[D.6.1] Holtec Report #HI-971717 to the USNRC; Evaluation of Haddam Neck Fuel Storage System under complete water loss scenario (1997).

[D.6.2] Holtec Report # HI-992135 to the USNRC; Evaluation of Millstone Point Unit 1 pool under complete loss of water (1999).

[D.6.3] Holtec Report #HI-2022858 to the USNRC; Thermal Evaluation of the potential of Zirc Fire in the Vermont Yankee pool under loss of water assumption (2002).

[D.6.4] ANSYS FLUENT Commercial Software, Lebanon, NH.

[D.6.5] Holtec Report #HI-992252 to the USNRC, Topical Report on the Validation of FLUENT (1999).

[D.6.6] Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, February 2001.

[D.6.7] HI-STORM 100 Safety Analysis Report, Holtec Report No. HI-2002444, Latest Revision.

[D.6.8] HI-STORM FW Safety Analysis Report, Holtec Report No. HI-2114830, Latest Revision.

[D.6.9] "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses," EPRI NP-5128, (April 1987).

[D.6.10] "Commercial Grade Dedication of FLUENT v18.1", Holtec Report HI-2177807, Revision 4.

[D.6.11] "Benchmarking Report on Thermal Models and Methodologies", Holtec Report HI-2166981, Latest Revision.

[D.6.12] Evaluation of Axial Burnup and Void Profiles, Holtec Report HI-2167312, Revision 2.

[D.6.13] "Thermal-Hydraulic Experiments Using a Dry Cask Simulator," NUREG/CR-7250, October 2018.

[D.6.14] "Validation of a Computational Fluid Dynamics Method Using Vertical Dry Cask Simulator Data", NUREG-2238, June 2020.

[D.6.15] "Thermal Validation Analysis of HI-STORM 100 System", Holtec Report HI-2188214, Revision 1.

SECTION B

HOLTEC INTERNATIONAL

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Telephone (856) 797-0900

Fax (856) 797-0909

May 28, 2021

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 99902086 - HDI Spent Fuel Pool Heatup Calculation Methodology

Subject: Response to Request for Additional Information - Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report

References:

1. Letter from Holtec International to US NRC, "Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report," September 29, 2020 (ML20280A524)

2. US NRC Electronic Mail Request to Andrea Sterdis (HDI) "Formal Transmittal of the US NRC Requests for Additional Information for Holtec Topical Report HI-2200750 Revision 0, "Holtec Spent Fuel Pool Heat Up Calculation Methodology," March 31, 2021 (ML21077A102)

Dear Sir or Madam:

In Reference 1, Holtec Decommissioning International, LLC (HDI) submitted a Topical Report providing a methodology for calculating Spent Fuel Pool heat up for NRC review and approval. Holtec believes the methodology will be a large benefit in reducing zirconium fire risks in the spent fuel pool.

In Reference 2, the NRC transmitted a request for additional information (RAI) concerning the Topical Report. The following Enclosures to this letter provide a response to the NRC RAI.

Enclosure 1 (submitted separately via the BOX) provides a proprietary version of the RAI response. This enclosure contains information proprietary to Holtec and is therefore supported by an affidavit signed by Holtec which is provided in Enclosure 3.

Enclosure 2 provides a non-proprietary, redacted version of the RAI response.

If you have any questions, please contact me at 856-797-0900 ext. 3813.

Sincerely,

Andrea Sterdis

Digitally signed by Andrea Sterdis
DN: cn=Andrea Sterdis, c=US, o=Holtec Decommissioning International, ou=HDI, email=a.sterdis@holtec.com
Date: 2021.05.28 16:24:25 -04'00'

Andrea L. Sterdis
VP, Regulatory and Environmental Affairs
Holtec Decommissioning International

Holtec Technology Campus, One Holtec Blvd, Camden, NJ 08104
Telephone (856) 797-0900
Fax (856) 797-0909

Enclosures:

Enclosure 1 (SUBMITTED SEPARATELY) Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up Calculation Methodology Topical Report (~~Holtec Proprietary Withhold Information from Public Disclosure pursuant to 10 CFR 2.390~~)
Enclosure 2 Holtec Responses to Request for Additional Information concerning Spent Fuel Pool Heat Up Calculation Methodology Topical Report (Non-Proprietary)
Enclosure 3 Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure
Enclosure 4 RAI 02, Reference 2.1 "Effective Thermal Conductivity and Edge Conductance Model for a Spent-Fuel Assembly

cc:
Robert Lucas, NRC, NRR/DORL/LLPB
Dennis Morey, NRC, NRR/DORL/LLPB
Ekaterina Lenning, NRC, NRR/DORL/LLPB
Christopher Regan, NRC, NMSS/DFM

Enclosure 2

Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up

Calculation Methodology Topical Report

Redacted Version

(19 Pages Attached)

RAI-01

Treatment of near-wall locations

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] **4.a, 4.b**

Regulatory Justification	SRP Section 15.0.2, Subsection III.3c				
Associated Section	3.1.4 Initial and Boundary Conditions				
Level of Concern	2	Level of Impact	5	Level of Effort	3
Overall Significance	Medium				
Holtec Response	**[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b** Therefore, the methodology proposed by Holtec does improve the safety of the spent fuel storage.				

RAI-02

<table>
<tr><td colspan="6">Lumped Analysis vs. Pin by Pin Analysis

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b</td></tr>
<tr><td>Regulatory Justification</td><td colspan="5">SRP Section 15.0.2, Subsection III.3b</td></tr>
<tr><td>Associated Section</td><td colspan="5">3.3.1.4 Level of Detail in the Model</td></tr>
<tr><td>Level of Concern</td><td>1</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td>1</td></tr>
<tr><td>Overall Significance</td><td colspan="5">High</td></tr>
<tr><td>Holtec Response</td><td colspan="5">[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b

The methodology in the main part of the report will be expanded as follows:

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

o</td></tr>
</table>

	[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390] **]** **4.a, 4.b** o The paper shows that the method is validated through measurements, **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b** • The approach is considered conservative due to the following reasons: **[**

	[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390] **]** [4.a, 4.b] This is further discussed in the response to RAI-03.

	Reference for Response to RAI-02. [2.1] Manteufel, R.D. and N.E. Todreas, “Effective Thermal Conductivity and Edge Configuration Model for Spent Fuel Assembly,” Nuclear Technology, Vol. 105, pp. 421–440, March 1994.

RAI-03

Radial and Axial Peaking **[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b**					
Regulatory Justification	SRP Section 15.0.2, Subsection III.3b				
Associated Section	3.3.1.4 Level of Detail in the Model				
Level of Concern	2	Level of Impact	2	Level of Effort	1
Overall Significance	High				
Holtec Response	**[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b** (see for example the HI-STAR 100 Storage SAR [3.1],				

	tables 2.1.3 and 2.1.4, and the SER on the initial submittal of this SAR [3.2], Section 4.3). **[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b** For PWR fuel, axial burnups were also extensively analyzed in support of Burnup Credit for spent fuel transportation casks, as documented in NUREG/CR-6801 [3.3], with results documented in Table 5 of that document. **[Proprietary Information Withheld Per 10 CFR 2.390.]** **4.a, 4.b** Note that for lower burnup, the NUREG reports slightly higher values, up to about 1.215 (for burnups between 14 and 18 GWd/mtU). **[Proprietary Information Withheld Per 10 CFR 2.390.]** **4.a, 4.b** For BWR fuel, similar studies were performed and are documented in NUREG/CR-7224 [3.4]. For high burnup assemblies, results are shown in that NUREG in Figure 6.3, with maximum values generally no more than 1.2. **[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** .

	[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

	[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] **4.a, 4.b**

References for Response to RAI-03

[3.1] HI-STAR 100 Final Safety Analysis Report, Holtec Report HI-2012610, Rev. 0, March 2001

[3.2] NRC Safety Evaluation Report and CoC, Holtec HI-STAR 100 Cask System, April 1999

[3.3] "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses", NUREG/CR-6801, ORNL/TM–2001/273, Oak Ridge National Laboratory, March 2003.

	[3.4] "Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit", NUREG/CR-7224, ORNL/TM-2015/544, Oak Ridge National Laboratory, August 2016. [3.5] "Horizontal Burnup Gradient Datafile for PWR Assemblies", DOE/RW-0496, Office of Civilian Radioactive Waste Management, May 1997. [3.6] Westinghouse Technology Systems Manual, Section 2.2, Power Distribution Limits, USNRC HRTD, Rev. 0508, ML11223A208

RAI-04

<table>
<tr><td colspan="6">

Time Step Sensitivity

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b

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<tr><td>Associated Section</td><td colspan="5">3.3.2.1 Numerical Solutions & 3.3.5.4 Sensitivity Studies</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td>4</td></tr>
<tr><td>Overall Significance</td><td colspan="5">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="5">

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b

[

] 4.a, 4.b

</td></tr>
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RAI-05

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<tr><td colspan="7">Planar Surface Area

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b</td></tr>
<tr><td>Regulatory Justification</td><td colspan="6">SRP Section 15.0.2, Subsection III.3e</td></tr>
<tr><td>Associated Section</td><td colspan="6">3.3.5.1 Important Sources of Uncertainty</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td colspan="2">4</td></tr>
<tr><td>Overall Significance</td><td colspan="6">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="6">[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b</td></tr>
</table>

RAI-06

<table>
<tr><td colspan="6">Uncertainty due to emissivity

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

]4.a, 4.b</td></tr>
<tr><td>Regulatory Justification</td><td colspan="5">SRP Section 15.0.2, Subsection III.3e</td></tr>
<tr><td>Associated Section</td><td colspan="5">3.3.5.1 Important sources of Uncertainty</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td>3</td></tr>
<tr><td>Overall Significance</td><td colspan="5">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="5">Surface emissivities are significantly affected by surface layers on the cladding (crud usually increases emissivity); therefore, the assumed oxidation layer and any exposed zircaloy surfaces are assumed to have the emissivity resulting from MATPRO Equation 4.1-8 [2] (equal to 0.8 or higher) using the oxidation thicknesses from [1]. Furthermore, Table B-3.II of [2] also shows an emissivity of fuel cladding with crud well over 0.8. Therefore, use of an emissivity of 0.8 for zircaloy cladding is conservative.

Emissivity of stainless-steel plates that are used for the rack cell walls is 0.587 per ORNL studies [3] and [4]. The variation in emissivity of stainless-steel with temperature is extremely small (~ 0.05) in large temperature range as shown in reference [5].

Moreover, it must be noted that the emissivity values of 0.8 and 0.587 for zircaloy cladding and stainless-steel plates, respectively, have been approved by USNRC in multiple Holtec's dry storage applications (USNRC Docket Nos. 72-1014, 72-1032, 72-1040, 71-9325, 71-9367, 71-9373, 71-9374, etc.). NRC staff further mentions in their SERs (Section 3.2 on Docket Nos. 71-9367, 71-9374) that the material properties and surface emissivities used in these applications are acceptable.</td></tr>
</table>

	The variances in emissivity can alter the radiation heat transfer characteristics of the surfaces and therefore change the peak cladding temperatures. However, as noted in Section 4.2.7 of [1], the impact of emissivity variations on the peak cladding temperature (PCT) is extremely small. As a defense-in-depth, Holtec also performed sensitivity evaluations **[** **[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]** **]** **4.a, 4.b** <u>References:</u> [1] "Spent Nuclear Fuel Effective Thermal Conductivity Report," US DOE Report BBA000000-01717-5705-00010 REV 0, (July 11, 1996). [2] Hagrman, Reymann and Mason, "MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, Tree 1280, Rev. 2, EG&G Idaho, August 1981. [3] "Nuclear Systems Materials Handbook, Vol. 1, Design Data", ORNL TID 26666. [4] "Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year-Old PWR Spent Fuel", ORNL/CSD/TM-149 TTC-0316, (1983). [5] "Process Heat Transfer", D.Q. Kern.

RAI-07

Quality Assurance Program

In the topical report, Holtec did not discuss the quality assurance program which controlled this analysis. Holtec should confirm that this **[Proprietary Information Withheld per 10 CFR 2.390]** [4.a, 4.b] analysis is kept under a quality assurance program consistent with 10 CFR Part 50 Appendix B that this program contains adequate documentation for design control, document control, software configuration control and testing, and corrective actions, and that the analysis has been independently peer reviewed. Additionally, Holtec should confirm that the important references which the analysis method rely upon have been incorporated into Holtec's quality assurance program.

Regulatory Justification	SRP Section 15.0.2, Subsection III.3f				
Associated Section	3.3.6.1 Appendix B Quality Assurance Program				
Level of Concern	3	Level of Impact	3	Level of Effort	2
Overall Significance	Low				
Holtec Response	The analysis developed for this topical report was developed, reviewed and approved under the Holtec Quality Assurance (QA) Program. The Holtec QA Assurance Program addresses the 10 CFR 50, Appendix B requirements and provides for appropriate design control, document control, software configuration control and testing, and corrective actions. The topical report and the supporting analysis are maintained under the Holtec QA Program. When the methodology is approved and then is applied to a plant specific spent fuel pool, the site specific calculations will be performed in accordance with the site's Quality Assurance Program.				

RAI-08

Comparison to Office of Research (RES) Data					
[[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]] 4.a, 4.b Please provide a plot similar to that given in Figure 7.3 with these comparisons					
Regulatory Justification	SRP Section 15.0.2, Subsection III.3d				
Associated Section	3.3.3.2 Validation of the Evaluation Model				
Level of Concern	3	Level of Impact	5	Level of Effort	3
Overall Significance	Low				
Holtec Response	Figure 7.3 in the TR has been expanded to show data up to 6 months for BWR fuel assemblies. The revised figure compares data from the method proposed in the TR to data from calculations done by the Office of Research (RES) starting from 6 months of cooling time. The conclusions made in the TR still remain applicable that the proposed method shows conservative results under all configurations for BWR fuel assemblies **[Proprietary Information Withheld per 10 CFR 2.390.]** 4.a, 4.b A similar figure has been added to the TR for PWR fuel assemblies.				

Response to RAI-08

Revised Figure 7.3

4.a, 4.b

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

RAI-09

<table>
<tr><td colspan="6">Variation in Heat Capacity

[

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b</td></tr>
<tr><td>Regulatory Justification</td><td colspan="5">SRP Section 15.0.2, Subsection III.3b</td></tr>
<tr><td>Associated Section</td><td colspan="5">3.3.1.4 Level of Detail in the Model</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td>4</td></tr>
<tr><td>Overall Significance</td><td colspan="5">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="5">Heat capacity of fuel assemblies is an input to the calculations. [

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

] 4.a, 4.b</td></tr>
</table>

Enclosure 4

Reference 2.1 (RAI Response RAI-02)

“Effective Thermal Conductivity and Edge Conductance

Model for a Spent-Fuel Assembly”

Copyrighted Material

SECTION C

HOLTEC INTERNATIONAL
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Fax (856) 797-0909

August 16, 2021

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 99902086 - HDI Spent Fuel Pool Heatup Calculation Methodology

Subject: Revised Response to Request for Additional Information - Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report

References:

1. Letter from Holtec International to US NRC, "Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report," September 29, 2020 (ML20280A524)

2. US NRC Electronic Mail Request to Andrea Sterdis (HDI) "Formal Transmittal of the US NRC Requests for Additional Information for Holtec Topical Report HI-2200750 Revision 0, "Holtec Spent Fuel Pool Heat Up Calculation Methodology," March 31, 2021 (ML21077A102)

3. Letter from Holtec International to US NRC, "Response to Request for Additional Information—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report," May 28, 2021.

Dear Sir or Madam:

In Reference 1, Holtec submitted a Topical Report providing a methodology for calculating Spent Fuel Pool heat up for NRC review and approval. Holtec believes the methodology will be a large benefit in reducing zirconium fire risks in the spent fuel pool.

In Reference 2, the NRC transmitted a request for additional information (RAI) concerning the Topical Report. The following Enclosures to this letter provide a response to the NRC RAI.

Reference 3 provided the Holtec response to the RAI that were made in Reference 2.

This letter provides a revised set of responses to the RAI that provides additional clarification and supporting information. Although the revisions are limited to the responses to RAI 02, 03, 05 and 08, for ease of review and tracking, HDI is providing complete revisions of the Reference 3 Enclosure 1 and 2. NRC should use Enclosures 1 and 2 in lieu of the Reference 3 Enclosures 1 and 2.

Enclosure 1 provides the proprietary version of the revised RAI response. This enclosure contains information

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proprietary to Holtec and is therefore supported by an affidavit signed by Holtec which is provided in Enclosure 3.

Enclosure 2 provides a non-proprietary, redacted version of the revised RAI response.

If you have any questions, please contact me at 856-797-0900 ext. 3813.

Sincerely,

Andrea Sterdis

Digitally signed by Andrea Sterdis
DN: cn=Andrea Sterdis, c=US, o=Holtec Decommissioning International, ou=HDI, email=a.sterdis@holtec.com
Date: 2021.08.16 20:09:32 -04'00'

Andrea L. Sterdis
VP, Regulatory and Environmental Affairs
Holtec Decommissioning International

Enclosures:

Enclosure 1 Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up Calculation Methodology Topical Report (Holtec Proprietary Withhold Information from Public Disclosure pursuant to 10 CFR 2.390)

Enclosure 2 Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up Calculation Methodology Topical Report—Redacted (Non-Proprietary)

Enclosure 3 Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure

cc:
Robert Lucas, NRC, NRR/DORL/LLPB
Dennis Morey, NRC, NRR/DORL/LLPB
Ekaterina Lenning, NRC, NRR/DORL/LLPB
Christopher Regan, NRC, NMSS/DFM

~~HOLTEC PROPRIETARY INFORMATION~~

Enclosure 2

Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up Calculation Methodology Topical Report

Proprietary Version Submitted Separately

Non-Proprietary Version (Redacted)

~~Withhold Information From Public Disclosure Under 10 CFR 2.390~~

(27 Pages not including this Cover Page)

RAI-01

Treatment of near-wall locations

[

.] [4.a, 4.b]

Regulatory Justification	SRP Section 15.0.2, Subsection III.3c				
Associated Section	3.1.4 Initial and Boundary Conditions				
Level of Concern	2	Level of Impact	5	Level of Effort	3
Overall Significance	Medium				
Holtec Response	[.] [4.a, 4.b] Therefore, the methodology proposed by Holtec does improve the safety of the spent fuel storage.				

RAI-02

Lumped Analysis vs. Pin by Pin Analysis

[

] 4.a, 4.b

Regulatory Justification	SRP Section 15.0.2, Subsection III.3b				
Associated Section	3.3.1.4 Level of Detail in the Model				
Level of Concern	1	Level of Impact	3	Level of Effort	1
Overall Significance	High				
Holtec Response	[ ] 4.a, 4.b For that, the methodology is expanded as follows: [				

	■ .] 4.a, 4.b

	○ .] 4.a, 4.b • The approach is considered conservative due to the following reasons: [

] 4.a, 4.b

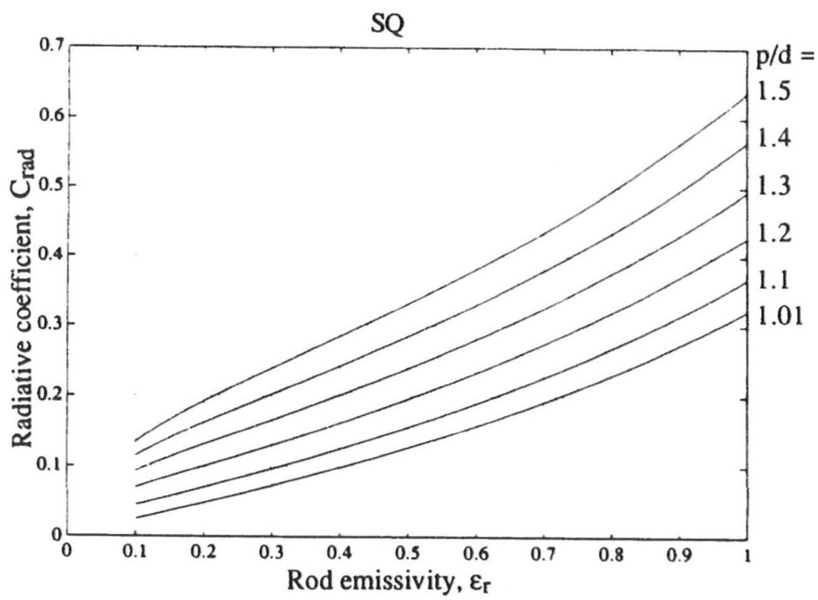


Figure 2-1: Radiative coefficient for a square array, Figure 3.6-5 in [2.2]

	References for Response to RAI-02 [2.1] Manteufel, R.D. and N.E. Todreas, "Effective Thermal Conductivity and Edge Configuration Model for Spent Fuel Assembly," Nuclear Technology, Vol. 105, pp. 421–440, March 1994. [2.2] Manteufel, R.D., "Heat Transfer in an Enclosed Rod Array", Submitted to the Department of Mechanical Engineering in partial fulfillment of the requirements for the degree of Doctor of Philosophy, Massachusetts Institute of Technology, May 1991

RAI-03

Radial and Axial Peaking

[

] 4.a, 4.b

Regulatory Justification	SRP Section 15.0.2, Subsection III.3b				
Associated Section	3.3.1.4 Level of Detail in the Model				
Level of Concern	2	Level of Impact	2	Level of Effort	1
Overall Significance	High				
Holtec Response	[ ] 4.a, 4.b (see for example the HI-STAR 100 Storage SAR [3.1], Table 2.1.8, and the SER on the initial submittal of this SAR [3.2], Section 4.3). [				

	•] 4.a, 4.b For PWR fuel, axial burnups were also extensively analyzed in support of Burnup Credit for spent fuel transportation casks, as documented in NUREG/CR-6801 [3.3], with results documented in Table 5 of that document. [.] 4.a, 4.b Note that for lower burnup, the NUREG reports slightly higher values, up to about 1.215 (for burnups between 14 and 18 GWd/mtU). [ ] 4.a, 4.b For BWR fuel, similar studies were performed and are documented in NUREG/CR-7224 [3.4]. For high burnup assemblies, results are shown in that NUREG in Figure 6.3, with maximum values generally no more than 1.2. [

	] 4.a, 4.b References for Response to RAI-03 [3.1] HI-STAR 100 Final Safety Analysis Report, Holtec Report HI-2012610, Rev. 0, March 2001 [3.2] NRC Safety Evaluation Report and CoC, Holtec HI-STAR 100 Cask System, April 1999

	[3.3] "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses", NUREG/CR-6801, ORNL/TM–2001/273, Oak Ridge National Laboratory, March 2003. [3.4] "Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit", NUREG/CR-7224, ORNL/TM-2015/544, Oak Ridge National Laboratory, August 2016. [3.5] Not used. [3.6] Westinghouse Technology Systems Manual, Section 2.2, Power Distribution Limits, USNRC HRTD, Rev. 0508, ML11223A208 [3.7] Safety Analysis Report on the HI-STAR 190 Package, Holtec Report HI-2146214, Rev. 3, November 2018, USNRC Docket 71-9378

RAI-04

Time Step Sensitivity

[

] 4.a, 4.b

Regulatory Justification	SRP Section 15.0.2, Subsection III.3d, Appendix K to 10 CFR 50, and *TMI {Three Mile Island] action items for PWR*				
Associated Section	3.3.2.1 Numerical Solutions & 3.3.5.4 Sensitivity Studies				
Level of Concern	3	Level of Impact	3	Level of Effort	4
Overall Significance	Low				
Holtec Response	[.] 4.a, 4.b				

[

] 4.a, 4.b

RAI-05

Planar Surface Area

[

] [4.a, 4.b]

Regulatory Justification	SRP Section 15.0.2, Subsection III.3e				
Associated Section	3.3.5.1 Important Sources of Uncertainty				
Level of Concern	3	Level of Impact	3	Level of Effort	4
Overall Significance	Low				
Holtec Response	[ ] [4.a, 4.b] [5.1] "Handbook of Heat Transfer", W.M. Rohsenow, J.P. Hartnett, Y.I. Cho, 3rd Edition, McGraw Hill Book Co. Extract from [5.1] on following page				

3. Infinite plane to row of parallel cylinders, or n rows of inline cylinders

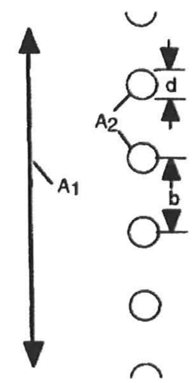
References: Hottel [290]; Kuroda and Munakata [296]

Definition: $D = d/b$

Governing equations:

For $n = 1$: $F_{1-2} = 1 - (1 - D^2)^{1/2} + D \tan^{-1}\left(\frac{1 - D^2}{D^2}\right)^{1/2}$

For $n > 1$: $F_{1-n\ \text{rows}} = 1 - (1 - F_{1-2})^n$



RAI-06

<table>
<tr><td colspan="6">Uncertainty due to emissivity

[

][4.a, 4.b]</td></tr>
<tr><td>Regulatory Justification</td><td colspan="5">SRP Section 15.0.2, Subsection III.3e</td></tr>
<tr><td>Associated Section</td><td colspan="5">3.3.5.1 Important sources of Uncertainty</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>3</td><td>Level of Effort</td><td>3</td></tr>
<tr><td>Overall Significance</td><td colspan="5">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="5">Surface emissivities are significantly affected by surface layers on the cladding (crud usually increases emissivity); therefore, the assumed oxidation layer and any exposed zircaloy surfaces are assumed to have the emissivity resulting from MATPRO Equation 4.1-8 [2] (equal to 0.8 or higher) using the oxidation thicknesses from [1]. Furthermore, Table B-3.II of [2] also shows an emissivity of fuel cladding with crud well over 0.8. Therefore, use of an emissivity of 0.8 for zircaloy cladding is conservative.

Emissivity of stainless-steel plates that are used for the rack cell walls is 0.587 per ORNL studies [3] and [4]. The variation in emissivity of stainless-steel with temperature is extremely small (~ 0.05) in large temperature range as shown in reference [5].

Moreover, it must be noted that the emissivity values of 0.8 and 0.587 for zircaloy cladding and stainless-steel plates, respectively, have been approved by USNRC in multiple Holtec's dry storage applications (USNRC Docket Nos. 72-1014, 72-1032, 72-1040, 71-9325, 71-9367, 71-9373, 71-9374, etc.). NRC staff further mentions in their SERs (Section 3.2 on Docket Nos. 71-9367, 71-9374) that the material properties and surface emissivities used in these applications are acceptable.</td></tr>
</table>

	The variances in emissivity can alter the radiation heat transfer characteristics of the surfaces and therefore change the peak cladding temperatures. However, as noted in Section 4.2.7 of [1], the impact of emissivity variations on the peak cladding temperature (PCT) is extremely small. As a defense-in-depth, Holtec also performed sensitivity evaluations [ ] 4.a, 4.b <u>Reference:</u> [1] "Spent Nuclear Fuel Effective Thermal Conductivity Report," US DOE Report BBA000000-01717-5705-00010 REV 0, (July 11, 1996). [2] Hagrman, Reymann and Mason, "MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, Tree 1280, Rev. 2, EG&G Idaho, August 1981. [3] "Nuclear Systems Materials Handbook, Vol. 1, Design Data", ORNL TID 26666. [4] "Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year-Old PWR Spent Fuel", ORNL/CSD/TM-149 TTC-0316, (1983). [5] "Process Heat Transfer", D.Q. Kern.

RAI-07

Quality Assurance Program

In the topical report, Holtec did not discuss the quality assurance program which controlled this analysis. Holtec should confirm that this [ ] [4.a, 4.b] analysis is kept under a quality assurance program consistent with 10 CFR Part 50 Appendix B that this program contains adequate documentation for design control, document control, software configuration control and testing, and corrective actions, and that the analysis has been independently peer reviewed. Additionally, Holtec should confirm that the important references which the analysis method rely upon have been incorporated into Holtec's quality assurance program.

Regulatory Justification	SRP Section 15.0.2, Subsection III.3f				
Associated Section	3.3.6.1 Appendix B Quality Assurance Program				
Level of Concern	3	Level of Impact	3	Level of Effort	2
Overall Significance	Low				
Holtec Response	The analysis developed for this topical report was developed, reviewed and approved under the Holtec Quality Assurance (QA) Program. The Holtec QA Assurance Program addresses the 10 CFR 50, Appendix B requirements and provides for appropriate design control, document control, software configuration control and testing, and corrective actions. The topical report and the supporting analysis as well as applicable references are maintained under the Holtec QA Program. When the methodology is approved and then is applied to a plant specific spent fuel pool, the site specific calculations will be performed in accordance with the site's Quality Assurance Program.				

RAI-08

<table>
<tr><td colspan="6">Comparison to Office of Research (RES) Data

[

.] [4.a, 4.b] Please provide a plot similar to that given in Figure 7.3 with these comparisons</td></tr>
<tr><td>Regulatory Justification</td><td colspan="5">SRP Section 15.0.2, Subsection III.3d</td></tr>
<tr><td>Associated Section</td><td colspan="5">3.3.3.2 Validation of the Evaluation Model</td></tr>
<tr><td>Level of Concern</td><td>3</td><td>Level of Impact</td><td>5</td><td>Level of Effort</td><td>3</td></tr>
<tr><td>Overall Significance</td><td colspan="5">Low</td></tr>
<tr><td>Holtec Response</td><td colspan="5">Figure 7.3 in the TR has been expanded to show data up to 6 months for BWR fuel assemblies. The revised figure compares data from the method proposed in the TR to data from calculations done by the Office of Research (RES) starting from 6 months of cooling time. The conclusions made in the TR still remain applicable that the proposed method shows conservative results under all configurations for BWR fuel assemblies [ ] [4.a, 4.b] A similar figure has been added below for PWR fuel assemblies.</td></tr>
</table>

Response to RAI-08

Revised Figure 7.3 and added Figure 7.3(b)

4.a, 4.b

RAI-09

Variation in Heat Capacity [ ] 4.a, 4.b					
Regulatory Justification	SRP Section 15.0.2, Subsection III.3b				
Associated Section	3.3.1.4 Level of Detail in the Model				
Level of Concern	3	Level of Impact	3	Level of Effort	4
Overall Significance	Low				
Holtec Response	Heat capacity of fuel assemblies is an input to the calculations. [ ] 4.a, 4.b				

Revised Appendix B (see Response to RAI-02)

EXAMPLE CALCULATION [ ][4a., 4.b] AT EXAMPLE PLANT A (Page B1 of 3) 4.a, 4.b

Revised Appendix B (see Response to RAI-02) (Page B2 of 3) 4.a, 4.b

Revised Appendix B (see Response to RAI-02) (Page B3 of 3)

4.a, 4.b

SECTION D

HOLTEC INTERNATIONAL

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October 18, 2021

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 99902086 - HDI Spent Fuel Pool Heatup Calculation Methodology

Subject: Response to Request for Additional Information 10 - Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report

References:

1. Letter from Holtec International to US NRC, "Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report," September 29, 2020 (ML20280A524)
2. US NRC Electronic Mail Request to Andrea Sterdis (HDI) "Formal Transmittal of the US NRC Request for Additional Information 10 for Holtec Topical Report HI-2200750 Revision 0, "Holtec Spent Fuel Pool Heat Up Calculation Methodology," October 14, 2021

Dear Sir or Madam:

In Reference 1, Holtec Decommissioning International, LLC (HDI) submitted a Topical Report providing a methodology for calculating Spent Fuel Pool heat up for NRC review and approval. Holtec believes the methodology will be a large benefit in reducing zirconium fire risks in the spent fuel pool.

In Reference 2, the NRC transmitted a request for additional information (RAI) concerning the Topical Report. The following Enclosures to this letter provide a response to the NRC RAI.

This letter provides the Holtec response to the NRC RAI 10 provided in Reference 2.

Enclosure 1 provides a proprietary version of the Holtec response to RAI 10. This enclosure contains information proprietary to Holtec and is therefore supported by an affidavit signed by Holtec which is provided in Enclosure 3. The RAI and RAI response provided in Enclosure 1 provide markings identifying those portions that are proprietary.

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Enclosure 2 provides a non-proprietary, redacted version of the Topical Report including Supplement 2.

If you have any questions, please contact me at 856-797-0900 ext. 3813.

Sincerely,

Andrea Sterdis

Digitally signed by Andrea Sterdis
DN: cn=Andrea Sterdis, c=US, o=Holtec Decommissioning International, ou=HDI, email=a.sterdis@holtec.com
Date: 2021.10.18 12:44:34 -04'00'

Andrea L. Sterdis
VP, Regulatory and Environmental Affairs
Holtec Decommissioning International

Enclosures:

Enclosure 1 Holtec Topical Report HI-2200750, Revision 2, "Holtec Spent Fuel Pool Heat Up Calculation Methodology." (Holtec Proprietary Withhold Information from Public Disclosure pursuant to 10 CFR 2.390)

Enclosure 2 Holtec Topical Report HI-2200750, Revision 2, "Holtec Spent Fuel Pool Heat Up Calculation Methodology." (Non-Proprietary)

Enclosure 3 Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure

cc:
Robert Lucas, NRC, NRR/DORL/LLPB
Dennis Morey, NRC, NRR/DORL/LLPB
Ekaterina Lenning, NRC, NRR/DORL/LLPB
Christopher Regan, NRC, NMSS/DFM

Enclosure 2

Holtec Response to Request for Additional Information concerning Spent Fuel Pool Heat Up

Calculation Methodology Topical Report

Non- Proprietary Version

(3 Pages not including this Cover Page)

RAI-10

[

] 4.a, 4.b

Holtec Response

Introduction

The Topical Report uses [ ] 4.a, 4.b to determine heat load limits, [

] 4.a, 4.b the analysis for a single rack cell surrounded by adiabatic boundaries in all direction. It is [ ] 4.a, 4.b to the adiabatic methodology discussed in References [1] and [3] in the Topical Report, which has been reviewed and approved by USNRC for several plants[1]. Based on this, [

] 4.a, 4.b

Comparison of Methods

The calculation and method used for Kewaunee is documented in Calculation 2013-07050, Rev 2, which is included in ML13351A040, beginning on page 37 of the document.

The principal characteristics of the methodologies are summarized in the table below.

Characteristic of the Methodology	Topical Report	Calculation 2013-07050 R2
No cooling by air considered	[ ] 4.a, 4.b	Section 1.1

[1] Examples: Kewaunee (ML14261A223); Fort Calhoun (ML16356A578); Pilgrim (ML19142A043)

Temperature limit of 900 °C	[ ] 4.a, 4.b	Section 1.2, Section 4.1
10 Hour time limit	[ ] 4.a, 4.b	Section 1.2
Reference to NUREG-1738 and SECY-99-168	[ ] 4.a, 4.b	Section 1.2
Only zirconium and uranium considered, and only along the heated length	[ ] 4.a, 4.b	Section 4.2,4.3 and 4.4
Starting temperature	[ ] 4.a, 4.b	Section 5.4 – 32 °C
Heat-up time starts when pool is drained	[ ] 4.a, 4.b	Section 5.5

The information in the table above supports the conclusion that [ ] 4.a, 4.b

USNRC SER and Approval

The USNRC SER (ML14261A223) discusses the calculations and methodology in various places. The most relevant place is Page 19 which states the following (emphasis added):

> The NRC staff found the adiabatic heat-up calculation adequate to demonstrate that a time exceeding 10 hours would be available before a significant radiological release might occur following an accident leading to loss of SFP water with no air cooling. The adiabatic heat-up calculation is a simplified method for determining the minimum time available for deployment of mitigation equipment and, if necessary, implementation of protective actions by offsite authorities using a CEMP approach. ***The methodology used was suitably conservative to compensate for simplifications related to phenomena such as axial variation in heat generation*** and the potential acceleration of the temperature increase as exothermic zirconium oxidation begins at high temperatures. The conservatisms include discounting the time for the water to drain from the SFP and neglect of additional heat sinks and heat transfer mechanisms that would exist in scenarios involving loss of SFP water inventory, even in situations where cooling air flow would be blocked.

This discussion confirms the adequacy of the methodology, and specifically addresses using the heat load without any further adjustments such as axial decay heat variations.

Summary

The methodology in the Topical Report to determine [ ] 4.a, 4.b to that discussed in References [1] and [3] to the topical report (i.e., adiabatic approach), and has been reviewed and approved by USNRC for several plants, including Kewaunee [

] **[4.a, 4.b]**

SECTION E

U. S. NUCLEAR REGULATORY COMMISSION

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE TOPICAL REPORT HI-2200750, REVISION 0,

"HOLTEC SPENT FUEL POOL HEAT UP CALCULATION METHODOLOGY"

HOLTEC INTERNATIONAL

DOCKET: 99902086

EPID: L-2020-TOP-0056

Enclosure 1

TABLE OF CONTENTS

1. INTRODUCTION

On September 29, 2020 (later updated by a submittal on October 30, 2020), Holtec International (HI or Holtec) submitted Topical Report (TR) HI-2200750, Revision 0, "Holtec Spent Fuel Pool Heat Up Calculation Methodology," (Ref. 1) to the U.S. Nuclear Regulatory Commission (NRC). This TR provides the methodology for describing transient heat up of the assemblies in the spent fuel pool (SFP) following a drain down event in a permanently defueled nuclear power plant.

The complete list of correspondence between the NRC and Holtec is provided in Table 1 below. This includes request for additional information (RAI) questions, responses to the RAI questions and any other correspondence relevant to this review.

Table 1: List of Correspondence

Author	Document	Document Date	Reference
Holtec	Initial Submittal	September 29, 2020	N/A
Holtec	Revised Submittal	October 30, 2020	1
NRC	Completeness Determination	December 4, 2020	2
NRC	Proprietary Determination	December 4, 2020	3
NRC	Request for Additional Information – Round 1	March 31, 2021	4
Holtec	RAI Responses	May 28, 2021[1]	5
Holtec	Revised RAI Responses	August 16, 2021	6
NRC	Request for Additional Information – RAI-10	October 1, 2021	17
Holtec	RAI Responses – RAI-10	October 18, 2021	18

In regard to the staff RAI questions, general information for each RAI including its number, its topic, its associated safety evaluation (SE) section, and the reference(s) of its response are given in Table 2 below.

[1] Document placed in ADAMS on June 23, 2021.

<u>Table 2: List of RAI Questions</u>

Question	Subject	Section of SE	Reference of Response
RAI-01	Treatment of near-wall locations	3.1.4	5, 6
RAI-02	Lumped Analysis vs. Pin-by-Pin Analysis	3.3.1.4	5, 6
RAI-03	Radial and Axial Peaking	3.3.1.4	5, 6
RAI-04	Time Step Sensitivity	3.3.5.4	5, 6
RAI-05	Planar Surface Area	3.3.5.1	5, 6
RAI-06	Uncertainty due to emissivity	3.3.5.1	5, 6
RAI-07	Quality Assurance Program	3.3.6.1	5, 6
RAI-08	Comparison to Office of Research (RES) Data	3.3.3.2	5, 6
RAI-09	Variation in Heat Capacity	3.1.4	5, 6
RAI-10	Current NRC Approved Methodology	3.3.1.3	18

2. <u>REGULATORY EVALUATION</u>

2.1. Applicable Regulations

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," contains regulations that define requirements for SFPs. These requirements are further specified in the general design criteria (GDC) given in Appendix A to Part 50 including: GDC-61, "Fuel Storage and Handling and Radioactivity Control," GDC-2, "Design Bases for Protection Against Natural Phenomena," GDC-4, "Environmental and Dynamic Effects Design Bases," and GDC-63, "Monitoring Fuel and Waste Storage." Holtec's methodology is focused on demonstrating that these regulations are still satisfied following a drain down event in the SFP by demonstrating that the spent fuel temperature remains below a given threshold for a specified time frame.

2.2. Applicable Guidance

NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," provides a technical analysis of the SFP accident risk at decommissioning nuclear power plants (Ref. 7). In Appendix 1.B of that report, the NRC provides the background for the proposed acceptance criteria, which vary based on the conditions experienced by the fuel. The acceptance criterion applicable to the Holtec methodology is that the fuel temperature must not exceed 900 °C prior to 10 hours following the drain down event. Satisfying this acceptance criterion will ensure that any significant global fuel damage and substantial release of fission products will be avoided.

To guide the NRC staff in performing its review and assure review quality and uniformity, the NRC created NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) (Ref. 8). Regulatory guidance for the review of design basis accident evaluation methodologies is provided in Section 15.0.2 of the SRP, "Review of Transient and Accident Analysis Methods" (Ref. 9). Similar guidance is also set forth for the industry in Regulatory Guide 1.203, "Transient and Accident Analysis Methods," dated December 2005 (Ref. 10). While this guidance was specifically developed for reviewing licensing basis accident analysis, it has been noted (Ref. 11) that this guidance is widely applicable to modeling and simulation in general. Therefore, the NRC staff followed the

guidance presented in SRP Section 15.0.2 but has modified that guidance commensurate with the risk-significance of the Holtec simulations.

2.3. Acceptance Criteria

NUREG-1738 provides different acceptance criteria depending on the scenario considered. For the Holtec methodology, the applicable acceptance criterion (as stated in the NUREG and the Holtec submittal) is demonstrating that the spent fuel temperature remains below 900 °C for at least 10 hours following a complete drain down of the SFP. Ensuring this acceptance criterion is satisfied would demonstrate that self-sustained oxidation (that would cause a further increase in the SFP temperatures) will not occur. While Holtec's methodology **[** **]** the NRC staff finds that the acceptance criterion provided in NUREG-1738 remains applicable.

NUREG-1738 does not specify whether or not it is the peak or an averaged fuel temperature which must remain below 900 °C for at least 10 hours. The NRC staff considered understanding the acceptance criterion as average fuel temperature but determined such an understanding was not reasonable because the criterion was based on the cladding temperature that causes self-sustained oxidation. For the average fuel temperature to reach 900 °C, some portion of the fuel must have been above 900 °C for some time period. The portion of the fuel above 900 °C would be hot enough to experience self-sustained oxidation, which would lead to increased fuel temperatures. **[** **]** Because the NRC staff did not understand this acceptance criterion as applying to the average spent fuel temperature as consistent with the goal of the criterion, the staff concluded that the acceptance criterion applied to the peak predicted SFP temperature.

3. TECHNICAL EVALUATION

Holtec has submitted a calculational methodology for simulating the temperature following the drain down event of an SFP. SRP Section 15.0.2 directs the reviewer to examine the evaluation model (EM), which is defined as the calculational framework for evaluating the behavior of the SFP and includes the computer programs, mathematical models, assumptions, and procedures on how to treat the input and the output, as well as many other factors. Based on guidance from the SRP, this review is organized into three categories shown in Table 3.

Table 3: General Review Categories

General Review Categories	
3.1	Scenario Identification Process
3.2	Documentation
3.3	Evaluation Model Assessment

3.1. Scenario Identification Process

The scenario identification process is a structured process used to identify the key figures of merit or acceptance criteria for the modeled accident. It is also used to identify and rank the component and physical phenomena modeling requirements based on their: (a) importance to

acceptable modeling of the scenario and (b) impact on the figures of merit for the calculation. Table 4 provides the SRP review criteria topics and the sections providing the NRC staff's review.

Table 4: Review Categories of the Scenario Identification Process

Scenario Identification Process	
3.1.1	Structured Process
3.1.2	Scenario Progression
3.1.3	Phenomena Identification and Ranking
3.1.4	Initial and Boundary Conditions

3.1.1. Structured Process

Structured Process

The process used for scenario identification should be a structured process.

SRP Section 15.0.2, Subsection III.3c

The Holtec submittal is focused on analyzing a single identified scenario, the heat-up resulting from the drain down of an SFP. As the scenario is pre-defined, the NRC staff finds that this criterion has been satisfied.

3.1.2. Scenario Progression

Scenario Progression

The description of each scenario should provide a complete and accurate description of the scenario progression.

SRP Section 15.0.2, Subsection III.3c

Unlike many scenarios, the heat-up following an SFP drain down event is well understood as the fuel heats up without any water to act as a coolant. **[**

] Because Holtec has described the scenario progression and such progressions are well-understood, the NRC staff finds that this criterion has been satisfied.

3.1.3. Phenomena Identification and Ranking

Phenomena Identification and Ranking

The dominant physical phenomena influencing the outcome of the scenario should be correctly identified and ranked.

SRP Section 15.0.2, Subsection III.3c

In the submittal, Holtec identified the dominant physical phenomena and how those phenomena would be modeled. **[**

] Because these phenomena are the dominant phenomena in the scenario and modeling them would result in an accurate or conservative prediction of reality, the NRC staff finds that this criterion has been satisfied.

3.1.4. Initial and Boundary Conditions

Initial and Boundary Conditions

The description of each scenario should provide complete and accurate description of the initial and boundary conditions.

SRP Section 15.0.2, Subsection III.3c

Holtec has provided a description of the initial and boundary conditions of the scenario in its submittal. **[**

]

The Holtec analysis method is based on **[**

]

[

]

Because Holtec has defined the initial and boundary conditions consistent with the scenario being modeled, the NRC staff finds that this criterion has been satisfied.

3.2. Documentation

Generally, the documentation for an EM is the focus of seven different review categories. However, the NRC staff did not need such detailed documentation based on two considerations: (1) the simplicity of the Holtec methodology and (2) that Holtec provided sufficient information such that the NRC staff could independently re-create the analysis described in the Holtec methodology. Because the information provided by Holtec was sufficient for the NRC staff to re-create the Holtec analysis, the NRC staff finds that the documentation is sufficient to fully describe the Holtec EM.

3.3. Evaluation Model Assessment

SRP Section 15.0.2, Subsection III.3.b, contains eight review criteria for EMs. The review criteria topics and the subsections that provide the NRC staff's assessments are listed in Table 5.

<u>Table 5: Evaluation Model Assessment Categories</u>

Subsection	
3.3.1	Evaluation Model Applicability
3.3.2	Evaluation Model Verification
3.3.3	Evaluation Model Validation
3.3.4	Evaluation Model - Data Applicability
3.3.5	Evaluation Model – Uncertainty Analysis
3.3.6	Evaluation Model - Quality Assurance Program

3.3.1. Evaluation Model Applicability

3.3.1.1. Previously Reviewed and Accepted Codes and Models

> **Previously Reviewed and Accepted Codes and Models**
>
> *It should be determined if the mathematical modeling and computer codes used to analyze the transient or accident should have been previously reviewed and accepted. If so, the reviewer should confirm that any previous conditions and limitations remain satisfied.*
>
> SRP Section 15.0.2, Subsection III.3b

The previously approved methods for performing this analysis are addressed in Section 3.3.1.3, "Physical Modeling," of this SE. Therefore, this criterion is addressed elsewhere.

3.3.1.2. Single Version of the Evaluation Model

> **Single Version of the Evaluation Model**
>
> *All assessment cases should be performed with a single version of the evaluation model.*
>
> SRP Section 15.0.2, Subsection III.3d

Based on the information provided in the TR, the NRC staff confirmed that all assessment cases were performed with a single version of the EM. The NRC staff concludes that this criterion has been satisfied.

3.3.1.3. Physical Modeling

Physical Modeling
The physical modeling described in the theory manual and contained in the mathematical models should be adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.
SRP Section 15.0.2, Subsection III.3b

In general, modeling an SFP drain down is simulating the heat up of a spent fuel assembly as well as any heat transfer mechanisms which can reduce the heat of that assembly. Thus, one common method to perform this analysis is to model the heat up of the assembly using the assembly's decay heat, conservatively ignoring all possible physical mechanisms that would transfer heat from that assembly. **[**

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The NRC staff finds that the physical model and the overall methodology of Holtec's approach would result in a reasonable or conservative estimate of the PCT and is therefore acceptable in determining if the acceptance criteria were satisfied. Further discussion regarding specific aspects of Holtec's assumptions in its heat transfer models **[**

] are addressed in Section 3.3.1.4, "Level of Detail in the Model," of this SE. Because Holtec is conservatively ignoring multiple heat transfer mechanisms between assemblies, is conservatively ignoring multiple heat sinks, and is accurately or conservatively analyzing each bundle in the SFP, the NRC staff finds that this criterion has been satisfied.

3.3.1.4. Level of Detail in the Model

> **Level of Detail in the Model**
>
> *The level of detail in the model should be equivalent to or greater than the level of detail required to specify the answer to the problem of interest.*
>
> SRP Section 15.0.2, Subsection III.3b

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In view of the foregoing, and particularly considering the conditions and limitations imposed by the staff, the level of detail in the model is adequate to determine if an SFP configuration meets the acceptance criterion, so the NRC staff finds that this criterion has been satisfied.

3.3.1.5. Simplifying and Averaging Assumptions

Simplifying and Averaging Assumptions

The simplifying assumptions and assumptions used in the averaging procedure should be valid for the accident scenario under consideration.

SRP Section 15.0.2, Subsection III.3b

[

]

3.3.1.6. Equations and Derivations

Equations and Derivations
The equations and derivations should be correct.
SRP Section 15.0.2, Subsection III.3b

The main equation used in the Holtec method is the general equation for radiation between two surfaces. Because Holtec is using a form of the equation for radiative heat transfer that is based on first principals and no derivations are necessary, the NRC staff finds that this criterion has been satisfied.

3.3.1.7. Field Equations

Field Equations

The field equations of the evaluation model should be adequate to describe the set of physical phenomena that occur in the accident.

SRP Section 15.0.2, Subsection III.3b

The main equation used in the Holtec method is the general equation for radiation between two surfaces. Because Holtec is using a form of the equation for radiative heat transfer that is based on first principals, the NRC staff finds that this criterion has been satisfied.

3.3.1.8. Code Tuning

Code Tuning

All code options that are to be used in the accident simulation should be appropriate and should not be used merely for code tuning.

SRP Section 15.0.2, Subsection III.3d

Because the NRC did not observe the selection of any options that were chosen merely for code tuning, the NRC staff finds that this criterion has been satisfied.

3.3.2. Evaluation Model Verification

3.3.2.1. Numerical Solution

Numerical Solution

The numerical solution should conserve all important quantities.

SRP Section 15.0.2, Subsection III.3d

[

]

Because Holtec performed an analysis demonstrating that the time step size was appropriately chosen, the NRC staff finds that this criterion has been satisfied.

3.3.3. Evaluation Model Validation

3.3.3.1. Validation of the Closure Relationships

Validation of the Closure Relationships
The range of validity of the closure relationships should be specified and should be adequate to cover the range of conditions encountered in the accident scenario.
SRP Section 15.0.2, Subsection III.3b

Holtec's methodology models the radiative heat transfer between fuel assemblies. **[**

] The NRC staff finds that this criterion does not apply.

3.3.3.2. Validation of the Evaluation Model

Validation of the Evaluation Model
Integral test assessments must properly validate the predictions of the evaluation model for the full-size plant accident scenarios. This validation should cover all of the important code models and the full range of conditions encountered in the accident scenarios.
SRP Section 15.0.2, Subsection III.3d

[

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While validation data of the EM are generally required, there are circumstances where no such data exist, and the NRC staff can still determine that the simulations are credible. One such circumstance is an EM that simulates well-understood phenomena where the uncertainties are fully characterized and can be conservatively treated (one example is given in Ref. 20). Because the phenomena are well understood, because the equation used to model the phenomena accurately describes the physics (i.e., first principles) of radiative heat transfer, because the variables in the equation are known with a high degree of certainty, because it is conservative to ignore the phenomena that are not being modeled, because the assumptions made that reduce the complexity of the model are each conservative in nature, and because of the large number of conservatisms in the methodology, the NRC staff determined that validation data is not required for use of this methodology to calculate PCT in the event of a SFP drain down event. Therefore, the NRC staff finds that this criterion does not apply.

3.3.3.3. Range of Assessment

> **Range of Assessment**
>
> *All code closure relationships based in part on experimental data or more detailed calculations should be assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data.*
>
> SRP Section 15.0.2, Subsection III.3d

In general, there are no closure relationships in Holtec's methodology. **[**

] Because there are no closure relationships used in Holtec's methodology, the NRC staff finds that this criterion does not apply.

3.3.3.4. Compensating Errors

> **Compensating Errors**
>
> *The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors.*
>
> SRP Section 15.0.2, Subsection III.3d

Because there was no comparison to experimental data, the NRC staff finds that this criterion does not apply.

3.3.4. Evaluation Model - Data Applicability

3.3.4.1. Assessment Data

> **Assessment Data**
>
> *Published literature should be referred to for sources of assessment data for specific phenomena, accident scenarios, and plant types.*
>
> SRP Section 15.0.2, Subsection III.3d

Holtec was able to provide assessment data in comparison to predictions of analysis performed by RES. This analysis is discussed in Section 3.3.3.2, "Validation of the Evaluation Model," of this SE. Additionally, **[**

] In general, because the NRC staff did not base its acceptance on validation data, the NRC staff finds that this criterion does not apply.

3.3.4.2. Similarity and Scaling

> **Similarity and Scaling**
>
> *The similarity criteria and scaling rationales should be based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses. Scaling analyses should be conducted to ensure that the data and the models will be applicable to the full-scale analysis of the plant transient.*
>
> SRP Section 15.0.2, Subsection III.3b

Because there was no comparison to experimental data, the NRC staff finds that this criterion does not apply.

3.3.5. Evaluation Model – Uncertainty Analysis

3.3.5.1. Important Sources of Uncertainty

> **Important Sources of Uncertainty**
>
> *The accident scenario identification process should be used in identifying the important sources of uncertainty. Sources of calculation uncertainties should be addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions). To address these uncertainties, demonstrate that the combined code and application uncertainty should be less than the design margin for the safety parameter of interest in the calculation.*
>
> SRP Section 15.0.2, Subsection III.3e

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3.3.5.2. Experimental Uncertainty

Experimental Uncertainty
The uncertainties in the experimental data base should be addressed. Data sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment should not be used.
SRP Section 15.0.2, Subsection III.3e

Because there was no comparison to experimental data, the NRC staff finds that this criterion does not apply.

3.3.5.3. Calculated and Predicted Results

> **Calculated and Predicted Results**
>
> *For separate effects tests and integral effects tests, the differences between calculated results and experimental data for important phenomena should be quantified for bias and deviation.*
>
> SRP Section 15.0.2, Subsection III.3e

Because there was no comparison to experimental data, the NRC staff finds that this criterion does not apply.

3.3.5.4. Sensitivity Studies

> **Sensitivity Studies**
>
> *Assessments should be performed where applicable {specific test cases for LOCA to meet the requirements of Appendix K to 10 CFR Part 50 and TMI [Three Mile Island] action items for PWR small-break LOCA}.*
>
> SRP Section 15.0.2, Subsection III.3d
>
> *Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.*
>
> Appendix K to 10 CFR Part 50
>
> *A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.*
>
> *TMI [Three Mile Island] action items for PWR*

In RAI-04, the NRC staff requested Holtec perform a time step sensitivity study to ensure a decrease in time step would not greatly impact its method's results. In its response (Ref. 5, later modified by Ref. 6), Holtec provided results of that sensitivity study demonstrating that a change in time step would not impact the results of its method. Various other sensitivity studies were also requested and provided in connection with other portions of this SE and have been fully addressed elsewhere. Because Holtec has provided all requested sensitivity studies, this criterion has been satisfied.

3.3.6. Evaluation Model - Quality Assurance Program

The Holtec Quality Assurance Program (QAP) covers, in part, the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the Holtec EM. The QAP also

ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

SRP Section 15.0.2, Subsection III.3.f, contains three review criteria for the QAP. The review criteria topics and the subsection providing the NRC staff's reviews are listed in Table 6.

Table 6: Quality Assurance Plan Review Categories

Subsection	
3.3.6.1	Appendix B Quality Assurance Program
3.3.6.2	Quality Assurance Documentation
3.3.6.3	Independent Peer Review

3.3.6.1. Appendix B Quality Assurance Program

Appendix B Quality Assurance Program

The evaluation model should be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

SRP Section 15.0.2, Subsection III.3f

In its TR, Holtec made reference to a number of reports (including NRC reports) that contained information important for Holtec's EM (e.g., material properties for UO_2), but which were not generated or maintained under an Appendix B QAP. In response to RAI-07 (Ref. 6), Holtec confirmed that it performed this analysis under its QAP and will also perform any plant specific SFP analysis under the licensee's QAP. Because Holtec has confirmed that the analysis will be maintained under a QAP that satisfies the requirements of 10 CFR Part 50 Appendix B, the NRC staff finds that this criterion has been satisfied.

3.3.6.2. Quality Assurance Documentation

Quality Assurance Documentation

The quality assurance program documentation should include procedures that address all of these areas [design control, document control, software configuration control and testing, and corrective actions].

SRP Section 15.0.2, Subsection III.3f

Due to the simplicity of the Holtec EM, this criterion is addressed under Section 3.3.6.1, "Appendix B Quality Assurance Program," above.

3.3.6.3. Independent Peer Review

Independent Peer Review
Independent peer reviews should be performed at key steps in the evaluation model development process.
SRP Section 15.0.2, Subsection III.3f

Due to the simplicity of the Holtec EM, this criterion is addressed under Section 3.3.6.1, “Appendix B Quality Assurance Program,” above.

3.4. Conclusions

The NRC staff has made the following conclusions based on the referenced evaluations provided in this SE:

- Based on the staff’s evaluation in Section 3.1, “Scenario Identification Process,” the NRC staff has determined that that the accident scenario identification process is a structured process and has been appropriately used to identify the key figures of merit for the SFP drain down event.

- Based on the staff’s evaluation in Section 3.2, “Documentation,” the NRC staff has determined that the documentation provided was sufficient to adequately describe Holtec’s methodology for performing the analysis of the SFP drain down event.

- Based on the staff’s evaluation in Section 3.3, “Evaluation Model Assessment,” the NRC staff has determined that:
  - the EM generated is applicable to the scenario,
  - Holtec has performed adequate verification analysis for the model,
  - based on the staff’s engineering judgment the model does not need additional validation analysis,
  - the model has had adequate uncertainty analysis performed, and
  - the QAP covers all relevant actions in the development and maintenance of the EM.

The NRC staff identified the following major uncertainties in the Holtec methodology:

- There is no experimental data to confirm the credibility of the simulations.

- There is limited data on the decay heat radial and axial peaking factors of the fuel in the SFP.

- It has not been mathematically proven that Holtec’s method will always result in analyzing the most limiting configuration.

- There is limited data on the applicability of **[ ]** at the temperatures of interest.

However, as discussed in this SE, the NRC staff concludes these uncertainties are more than offset by the following:

- Holtec is conservatively ignoring any heat transfer from the fuel assemblies to other structures in the SFP. This ensures that the modeled heat in the spent fuel assemblies will be conservatively higher than expected. This includes ignoring convection, conduction, and radiation to the plates separating the fuel assemblies, the SFP walls, and ultimately to the environment.

- Holtec is conservatively ignoring any natural convection from the air or steam that would be expected to flow through assemblies following an SFP drain down event. This ensures that the heat in the spent fuel assemblies will be conservatively higher than expected.

- Holtec is conservatively treating the fuel pellet and fuel cladding as a single material. This will result in a conservatively high cladding temperature, as there would normally be a temperature distribution in the fuel pin that would have its peak in the pellet and would decrease as the heat is transferred through the pellet, through the gap, and through the cladding. Holtec's method ignores this temperature gradient and calculates a single lumped clad/fuel temperature that represents the average temperature of the fuel pin and will always be higher than the cladding surface temperature where the acceptance criterion is applied.

- Holtec is conservatively ignoring the neutron absorbing material, which would act as a heat sink and reduce the temperature in the assembly.

- Holtec's analysis methodology is based on radiative heat transfer, which is well understood. Further, the NRC staff has concluded that as the temperatures increase, the method's predictions would become more conservative due to ignoring the impacts of an increasing surface emissivity.

- Holtec's analysis methodology has been confirmed using higher fidelity models.

- Holtec's **[ ]** methodology generally results in a conversative limiting decay heat power for fuel assemblies **[ ]**

Based on the above conclusions, the NRC staff finds that there is reasonable assurance that Holtec's EM (described in Ref. 1, with modifications provided by Refs. 5 and 6) along with the staff's conditions and limitations, conservatively or accurately predicts the PCT following an SFP drain down event, and use of the method is acceptable for ensuring that the acceptance criteria of 900℃ is satisfied.

3.4.1. Conditions and Limitations

1) The Holtec methodology is approved with the limitation that the decay heat values - including axial peaking factors - of all assemblies are analyzed in order to determine the most limiting case. **[ ]**

The following three conditions and limitations only apply in situations **[ ]**

[

]

2) The Holtec methodology is approved with the condition that the modified form of Equation (26) - as stated in response to RAI-02 and given in Equation (1) in this SE - is used to determine the PCT after applying the method described in Chapter 3 of the TR to determine the average fuel temperature. **[** **]**

3) The Holtec methodology is approved with the condition that a **[** **]** must be applied to the PCT.

4) The Holtec methodology is approved with the condition that **[** **]**

4. <u>REFERENCES</u>

1. Sterdis, A., Holtec Decommissioning International (HDI), Holtec International (HI), to NRC, HI-2200750 Revision 0, “Holtec Spent Fuel Pool Heat Up Calculation Methodology,” September 29, 2020 – initial submittal (October 30, 2020 – resubmittal) (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20280A525 (Non-Publicly Available) and ML20280A524 (Publicly Available)).

2. Lenning, E., NRC, email to A. Sterdis, HDI, HI, “Completeness determination for Holtec Decommissioning International Topical Report HI-2200750 Revision 0, ‘Holtec Spent Fuel Pool Heat Up Calculation Methodology’,” December 4, 2020 (ADAMS Accession No. ML20329A373 (Transmittal Email), ML20329A368 (Completeness Determination)).

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Date: March 25, 2022

Appendix A Description of Confirmatory Calculations

The purpose of this appendix is to describe the confirmatory calculations that were performed by the NRC staff in support of the review of Holtec International (HI) HI-2200750, Revision 0, (Hereafter referred to as HI-2200750). Two sets of calculations were performed. The first set of calculations was a direct re-creation of the calculational method described in HI-2200750. These calculations were initially performed to confirm NRC staff's understanding of the method. **[**

] These calculations are described in A.1 Recreation of HI-2200750 Method. The second set of calculations were **[**

] These calculations are described in A.2 Pin by Pin Simulation.

A.1 Re-creation of HI-2200750 Method

Initially, this model was developed to reproduce Holtec's method in order to verify completeness of description within the TR and facilitate reviewer understanding of the method. To that end, the equations and properties employed are effectively identical to that described in the TR. **[**

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A.2 Pin by Pin Simulation

These simulations were performed to **[**

] using ANSYS Fluent. **[**

] Gambit was used to generate the geometry and the mesh. **[**

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Parameters provided in HI-2200750 were used to facilitate comparison with Holtec's calculations. **[**

] were taken from the publicly available DOE/RW-0184 Vol. 3 (Ref. 1).

[

] Homogenized density and specific heat capacity were calculated using the following equations

$$\rho_{homogenized} = (m_{UO_2} + \rho_{zircaloy} V_{zircaloy})/(V_{UO_2} + V_{zircaloy})$$

$$c_{p,homogenized} = (m_{UO_2} c_{p,UO_2} + \rho_{zircaloy} V_{zircaloy} c_{p,zircaloy})/(m_{UO_2} + \rho_{zircaloy} V_{zircaloy})$$

Where ρ, c_p, V, and m represent density, specific heat capacity, volume in an assembly, or mass in an assembly, respectively, of the material named in the subscript. Zircaloy density was taken from Table 5.2, UO_2 mass from Table 5.3, and UO_2 specific heat from Appendix A of HI-2200750. Zircaloy specific heat capacity was calculated using equation 3-14 with the values listed in Table 5.1 of HI-2200750. Volumes were calculated using the dimensions in Table 5.3.

Specific heat capacity of the homogenized material was calculated at the same temperature intervals used in Appendix A of HI-2200750 and linearly interpolated between these intervals. Material density, dimensions, and **[ ]** were modeled as constant with respect to temperature.

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Manteufel discussed a similar consideration in Appendix A of his dissertation. Specifically, he evaluated a modified Biot number to determine whether the effect of conduction within a fuel pin needed to be considered. The Biot number Bi is the ratio of the thermal resistance to conduction within the body of an object to the thermal resistance to convection at its surface.

$$Bi = hL_c/k$$

Where h is the convective heat transfer coefficient, L_c is a characteristic length, and k is the thermal conductivity of the material. Smaller Biot numbers indicate that the thermal resistance due to conduction within the body is smaller, and the assumption of uniform temperature within the object is better. In Manteufel's application, the relevant comparison was between conduction heat transfer and radiative heat transfer, so Manteufel substituted the convective heat transfer coefficient with an approximated heat transfer coefficient due to radiation.

$$h_r = 4\sigma\epsilon T^3$$

For PWR applications, Manteufel calculated a Biot number of approximately 0.02 using the worst-case emissivity and thermal conductivity, indicating that the assumption of isothermal rods would not have a significant impact on the solution. However, the maximum temperature

considered in his application was 300 °C. In the present application, temperatures can reach 900 °C, so more energy will be transferred through radiation, and temperature gradients within the pin will have a greater impact on the solution. Biot numbers between 0.01 and 0.15 were calculated assuming a cladding temperature of 900 °C, depending on whether cladding or fuel properties and dimensions are used. The exact relationship between the magnitude of the Biot number and the **[** **]** is not clear, but generally a Biot number of less than 0.1 has been used to justify the isothermal temperature assumption. For the majority of the transient, the temperature will be lower than 900 °C, so the Biot number will be less limiting. Because the Biot number for this application remains relatively small, conduction within each pin was not simulated.

Simulated cases are shown in Table 7. In each case, the power density in each pin is determined by dividing the nominal assembly power by total volume of the fuel pins **[**

] Cases 7, 8, and 9 were run to confirm that the method Holtec proposed in response to RAI-02 would result in a conservative **[**

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Results for all cases are plotted in Figures 17 through 22 and Figures 24 through 26. Simulations were run for a fixed period of 20 hours to ensure that PCTs exceeded the limit during the simulation time. The limiting temperature, 1173.15 K, is plotted as a black dotted line. [

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ATTACHMENT 3

Affidavit Pursuant to
10 CFR 2.390

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Jean A. Fleming, being duly sworn, depose and state as follows:

1) I have reviewed the information described in paragraph (2) which is sought to be withheld and am authorized to apply for its withholding.

2) The information sought to be withheld is in Attachment 1 to the Enclosure to Holtec Letter 3301001. These documents information that is proprietary to Holtec International.

3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

4) Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

   b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b above.

5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or

AFFIDAVIT PURSUANT TO 10 CFR 2.390

other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive decommissioning and spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

   The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

   The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF CAMDEN)

Jean A. Fleming, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Camden, New Jersey, this 21st day of December 2022.

Jean A. Fleming

Jean A. Fleming
VP, Licensing, Regulatory Affairs and PSA
Holtec International

Subscribed and sworn before me this 21 day of December, 2022

LAUREN MADISON MURDOCH
NOTARY PUBLIC
STATE OF NEW JERSEY
MY COMMISSION EXPIRES AUGUST 13, 2027
COMMISSION: #50201915