September 3, 1999

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE L.L.C.	)	Docket No. 72-22-ISFSI
	)	
(Private Fuel Storage Facility)	)	

APPLICANT'S RESPONSE TO STATE OF UTAH'S REQUEST FOR ADMISSION OF LATE-FILED SECOND AMENDED UTAH CONTENTION Q

Applicant Private Fuel Storage L.L.C. ("Applicant" or "PFS") hereby responds to the "State of Utah's Request for Admission of Late-Filed Second Amended Utah Contention Q," filed August 20, 1999. ("State's 2nd Request"). Like its initial late-filed amended Contention Q[1], the State's Request should be denied, first, for failing to meet the requirements for late-filed contentions, and second, for failing to meet the Commission's contentions requirements set forth in 10 C.F.R. § 2.714.

I. BACKGROUND

As part of its June 1997 License Application, PFS included the results of its cask vendors' analyses of vertical drops and tipover events. See Safety Analysis Report ("SAR") at 8.2.6 (rev. 0). Based on the license application, the State filed a contention (Contention Q) which alleged, in part, that PFS did not adequately identify the "most vulnerable fuel" analyzed in a cask drop, and that PFS did not address lifting accidents. The Board rejected the contention in its entirety. Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation) LBP-98-7, 47 NRC 142, 195 (1998).

[1] See State of Utah's Request for Admission of Late-Filed Amended Utah Contention Q (July 22, 1999) (hereinafter "State's 1st Request").

In February 1998, the State's expert. Dr. Marvin Resnikoff, whose declaration supports the State's 2nd Request, began an exchange of letters with the NRC Staff concerning the methodology developed by the Lawrence Livermore National Laboratory ("LLNL")[2] for analyzing the impacts of a cask drop on fuel integrity, including the dynamic loading by fuel pellets and the effects of irradiated fuel cladding, the precise issues that underlie the State's 2nd Request (as well as its 1st Request).

On May 21, 1999, the NRC Staff issued Interim Staff Guidance 12 – Buckling of Irradiated Fuel Under Drop Conditions ("ISG-12"), which recommended that the analysis of cask drop accidents include consideration of the effects of irradiated fuel cladding and pellet weight. On July 22, 1999, the State filed its 1st Request alleging, based on ISG-12, that PFS was required to perform a revised analysis of fuel integrity for a vertical drop event incorporating pellet weight and irradiated fuel cladding, and had failed to do so. After PFS and the Staff pointed out that Holtec had performed such an analysis, the State withdrew its 1st Request.[3]

On August 20, 1999, the State filed its 2nd Request based on alleged inadequacies in the Holtec analysis. Specifically, the 2nd Request alleges that (1) Holtec failed to consider the combined effect of cladding embrittlement due to irradiation and the potential thinning of cladding for high burnup fuel, and (2) that Holtec's analysis did not consider "the dynamic effects of a cask drop accident." State's 2nd Request at 6.

[2] "Dynamic Impact Effects on Spent Fuel Assemblies", UCID-21246 (October 1987).

[3] State of Utah's Reply to Applicant's and NRC Staff's Responses to Amended Q and Notice of Withdrawal of Amended Contention Q (August 18, 1999).

II. ARGUMENT

The State's late-filed Amended Contention Q should not be admitted first, because it does not satisfy the NRC's requirements for late-filed contentions, and second, because it seeks to require PFS to perform an analysis that is properly within the scope of the rulemaking for Holtec's certificate of compliance. Moreover, the contention must be dismissed because it fails to present a genuine dispute of material fact.

A. The State's Request to File Amended Contention Q Is Unjustifiably Late

The State must demonstrate that a balancing of the five factors set forth in 10 C.F.R. § 2.714(a)(1)(i)-(v) supports admission of its late-filed contention, LBP-98-7, 47 NRC at 167, which it has failed to do. Hence, it request must be denied.

1. The State Lacks Good Cause

The first and most important factor in determining the admissibility of a late-filed contention is a showing of good cause. The State lacks good cause here because the bases for its contention have been available to the State for much longer than the period required by the Board for timely filing.[4]

Holtec's alleged failure to consider the effects of irradiation and pellet weight have been known to the State, through its expert Dr. Resnikoff, for at least 17 months before this contention was filed.[5] The State's familiarity with the concerns of cladding irradiation and the dynamic load from fuel pellets is evidenced by Dr. Resnikoff's dialogue with the Staff and the State's comments on the Holtec HI-STAR 100 storage cask. In Dr.

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[4] See Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation) LBP-99-3, 49 NRC 40, 47 (1999) (stating the 45 days approaches the limit for timeliness).

[5] Dr. Resnikoff copied his February 27, 1998 letter to Denise Chancellor, the State's Assistant Attorney General and Connie Nakahara of the Utah Division of Environmental Quality. See Letter from M. Resnikoff to C. Haughney, dated February 27, 1998 (attached as Exhibit 1).

Resnikoff's February 27, 1998 letter to the NRC, he specifically questions the LLNL methodology's use of "non-irradiated fuel assemblies" and its failure to "take into account the weight of the fuel itself." Exh. 1 at 2. When the Staff responded that it had evaluated his concerns for a horizontal drop accident,[6] Dr. Resnikoff again wrote the Staff, stating they "did not fully answer [his] concerns" and requested that they further evaluate "[his] concerns about brittleness" and the "important distinction between static and dynamic loading."[7] The State's prior knowledge of the concerns raised in the State's 2nd Request is further illustrated by its March 26, 1999 comments in the rulemaking for Holtec's HI-STAR 100 certificate of compliance.[8] In its comments, the State, with the assistance of Dr. Resnikoff, specifically questioned Holtec's reliance on the LLNL methodology, and the methodology's alleged failure to address the impacts of irradiated cladding and the dynamic loads from fuel pellets. Exh. 4 (State's Comments) at 2-6.

Neither is the third basis offered in the State's 2nd Request, the potential thinning of cladding for high burnup fuel, is not based on any new information contained in Holtec's revised analysis. Instead, the State cites to an NRC Information Notice released over one year ago. State's 2nd Request at 8 (referencing Information Notice 98-29, Predicted Increase in Fuel Rod Cladding Oxidation (August 3, 1998)). The State had actual knowledge of this Information Notice for at least five months before filing the 2nd Request, as the State referenced it in its March 26, 1999 comments on the HI-STAR 100 Storage cask. State's Comments at 3-4.

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6 Letter from M. Delligatti to M. Resnikoff, dated November 19, 1998 (attached as Exhibit 2).

7 Letter from M. Resnikoff to M. Delligatti, dated December 31, 1998 (attached as Exhibit 3).

8 Letter from D. Chancellor to Secretary, NRC, dated March 26, 1999 ("State's Comments") (attached as Exhibit 4).

Rather than explain how its contention is derived from the Holtec analysis and why it could not have been filed earlier, the State simply claims that good cause exists because PFS has yet to file a license amendment and "because [the State] has diligently pursued the issue of the inadequacy of the Applicant's cask stability analysis . . . ." State's 2nd Request at 13. These arguments are no excuse for the State's lack of timeliness.

When the license amendment discussing Holtec's analysis was filed[9] is irrelevant to the admissibility of this particular contention because the contention is not based on information that would be contained in the license amendment. Instead, the contention is based on the alleged failure by Holtec to consider certain additional factors that might influence the analysis. Thus, the State cannot now claim that their alleged absence from PFS's fifth license amendment somehow justifies the State's failure to raise these specific issues based on the original license application, or even the four previous license amendments.

Nor does the State's supposedly diligent pursuit of this issue through other means somehow justify its failure to file a timely contention. As the Commission has clearly determined, intervenors cannot simply wait to file a contention when the information supporting the contention has previously been publicly available.[10] The State has an "ironclad obligation to examine [on a timely basis] the publicly available documentary material . . . ."[11] Here, the information supporting the contention was not only publicly

[9] On August 27, 1999, PFS submitted License Amendment No. 5, which contains a discussion of the Holtec buckling analysis. See SAR § 8.2.6.2 (rev 5.0).

[10] See Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041, 1048 (1983).

[11] Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), LBP-83-8A, 17 NRC 282, 285 (1983).

available, but has been explicitly discussed for many months by the State and its expert. Its failure to fulfill this obligation cannot justify the admission of an untimely contention.

The State therefore lacks good cause. Where good cause is lacking, a compelling showing must be made on the other four factors, which the State has not done here.

2. The Other Factors Do Not Justify Admission of the Late-Filed Contention

Of the remaining four factors, the third and fifth factors are to be accorded more weight than the second and fourth factors, which concern the protection of the petitioner's asserted interest by other means or parties. LBP-98-7, 47 NRC at 207-209. While the State's interests may not be represented by another party in the PFS proceeding, it certainly has other means available to protect its interests, namely, the rulemaking associated with the certificate of compliance for the Holtec HI-STORM 100 storage cask.[12] As evidenced by its filing of comments for the rulemaking for the HI-STAR 100 storage canister, the State is well aware of the certificate of compliance rulemaking process and can represent its interests in those proceedings. See Exh. 4 (State's Comments).

The State's claim that the generic rulemaking process for storage casks does not provide an adequate means to protect its interests is a direct challenge to the Commission's regulations. As decided in Kelly v. Selin, 42 F.3d 1501 (6th Cir. 1995), the generic rulemaking process established by the NRC for certifying storage casks is a permissible exercise of the Commission's statutory authority, despite the lack of an adjudicatory hearing. The State's argument that "[the generic rulemaking process] is a very different

[12] The comment period for the HI-STORM cask has not yet opened but the Staff has issued a Preliminary Draft Safety Evaluation Report to Holtec and is expected to publish the Draft Safety Evaluation Report and a notice of opportunity for comment in the Federal Register this fall. See Proposed Schedule provided by NRC Staff at December 11, 1998 Pre-hearing Conference.

type of proceeding, which affords the State much less of an opportunity to vindicate its views." State's 2nd Request at 15, is remarkably similar to the arguments clearly rejected by the Sixth Circuit. A party's displeasure with the procedures of the proper forum is no justification for allowing admission of a contention in an inappropriate forum. Accordingly, this factor weighs against admitting the contention.

Likewise, neither the third nor the fifth factor support the State here. The State has not established Dr. Resnikoff's expertise to assist in developing a sound record for determining fuel cladding structural integrity for cask drop and tipover, particularly in view of his failure to recognize that Holtec, as shown below, considered the effects of irradiation in its analysis. Also, contrary to the State's assertion, admission of the contention will certainly broaden and inevitably delay this proceeding by expanding its scope to include a contention that has already been dismissed by the Board and thus is not the subject of any existing contention.

In sum, the remaining four factors weighed together militate against granting the State's late-filed motion, and therefore clearly do not make the compelling showing required to overcome the State's lack of good cause.

B. The State's Amended Contention is Inadmissible

In its basis for Amended Contention Q, the State claims that PFS's License Application is inadequate because Holtec's analysis of spent fuel integrity under the design basis vertical acceleration for the HI-STORM storage cask system does not consider the effects of cladding irradiation, cladding thinning, and dynamic loads from fuel pellets. State's 2nd Request at 6. The State's contention must be rejected because (1) the proper forum for raising concerns regarding the adequacy of Holtec's analysis of fuel integrity under design basis accelerations is the certificate of compliance rulemaking for the cask, and (2) the State's contention does not present any genuine dispute of material facts.

The State's contention is inadmissible in that it "impermissibly challenge[s] the Commission's regulatory scheme, provisions, or rulemaking-associated generic determinations, which establish a separate cask design approval process . . . ." LBP-98-7, 47 NRC at 186. As the Board has previously recognized, generic issues concerning the adequacy of the vendors' designs are to be addressed in the separate rulemaking proceedings for certification of the casks, not the licensing of the PFSF. Id.[13] The issue of the fuel assemblies' integrity under design basis drop conditions is a generic one, and the State has made no attempt to show how the conditions at PFS are unique. Thus, if the State has concerns with Holtec's analysis of fuel integrity under design bases accelerations for its casks, the proper forum for raising them is the rulemaking for the HI-STORM 100 certificate of compliance.[14] The State's attempt to raise this generic design issue as part of this proceeding is unwarranted and the Contention should be dismissed.

The State's contention must also be dismissed for failing to present any genuine dispute of material fact. First, the State's claim that Holtec's analysis fails to consider the effects of irradiation on the cladding, as recommended by ISG-12, is incorrect. In Revision 7.0 to the Topical Safety Analysis Report ("TSAR") for HI-STORM 100, Holtec includes a revised analysis of fuel integrity under drop conditions that incorporates the recommendations in ISG-12, including specifically, the use of irradiated fuel cladding mate-

[13] See also Private Fuel Storage, L.L.C., LBP-98-10, 47 NRC 288, 295 (1998).

[14] As noted above, both Dr. Resnikoff and the State have raised similar issues in context of the rulemaking proceeding for the HI-STAR 100 cask storage system. The Commission has rejected these arguments in the final rule adding HI-STAR 100 to the list of approved casks. 64 Fed. Reg. 48,259 (Sept. 3, 1999).

rials. See HI-STORM TSAR. Section 3.5 (Rev. 7.0) (attached as Exhibit 5). The TSAR specifically states that "[t]he material properties used in the non-linear analysis are those for irradiated Zircalloy . . . ." Id. at 3.5-3. The State provides no evidence to suggest the contrary. That Holtec's analysis complies with the Staff's concerns addressed in ISG-12 is further evidenced by the Staff's approval of Holtec's buckling analysis for the HI-STAR 100 storage and transportation system (64 Fed. Reg at 48,261-62) and its planned issuance of the draft Safety Evaluation Report for the HI-STORM storage cask.

Second, the State has failed to show any genuine issue of material fact concerning the alleged need to evaluate the dynamic loading from fuel pellets. Despite its knowledge of this issue for 17 months, the State and Dr. Resnikoff still offer no evidence beyond mere speculation that dynamic loading will have any significant effect on cladding integrity. Indeed, the State has provided no technical basis for its claim that fuel pellets traveling less than 0.1mm[15] would somehow cause a failure of cladding integrity, especially considering that the fuel cladding can resist a deceleration of 63.5 g's, which is 40% greater than the design basis acceleration of 45 g's. HI-STORM TSAR at 3.5-15-19.

Third, as shown by the State's own calculations, State's Exhibit 4, the State's claims concerning the potential thinning of cladding for high burnup fuel are not material to this proceeding. Even if the full 17% thinning occurred, buckling would only occur at 50.81 g's by the State's calculations, still above the maximum design load of 45 g's. Id. In any event, the State has failed to show why thinning is even an issue since the HI-STORM cask storage system is not presently certified to take the high burnup fuel that

[15] The gap between the cladding and fuel pellets for the Westinghouse 17x17 Fuel Assembly is 0.082 mm. See Henry Graves, Nuclear Fuel Management, Appendix C – Typical Nuclear Reactor Power Data, (1979) (attached as Exhibit 6).

would cause the increased thinning.[16] In short, the amended contention must also be dismissed for failing to present any genuine dispute of material fact.[17]

The State also contends, incorrectly, that the revised analysis must be performed for the Intermodal Transfer Point ("ITP") and "during transport on either rail or highway." State's Request at 7. As this Board decided in granting summary disposition for Utah Contention B, transportation of spent fuel is governed by 10 CFR Part 71, and not Part 72, and is beyond the scope of this proceeding. Private Fuel Storage, L.L.C., LBP-99-34, 50 NRC ___ (1999).Thus, this part of the State's contention must be rejected.

III. CONCLUSION

For the foregoing reasons, Applicant respectfully requests that the Board deny Utah's request to admit its late-filed, second amended Contention Q.

Respectfully submitted,

Jay E. Silberg

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Ernest L. Blake, Jr.
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September 3, 1999

[16] PFS plans to accept high burnup fuel in the future. Before this can occur, Holtec would have to amend an approved HI-STORM certificate of compliance, at which time the State could raise any concerns.

[17] The State's discussion of "the concept of multiple confinement," State's 2nd Request at 9-11, does not refute the authority cited at pages 209-210 in Applicant's December 24, 1997 Answer to Petitioner's Contentions, in particular the quotation from the proposed rule (51 Fed Reg. 19,106, 19,108 (1986)) which explicitly provides that the "canister could act as a replacement for the cladding." Thus, the fact that the State's contention would not entitle the State to relief even if it was proved true constitutes another basis for the dismissal of Second Amended Contention Q.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Commission

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE L.L.C.	)	Docket No. 72-22
	)	
(Private Fuel Storage Facility)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of Applicant's Response to State of Utah's Request for Admission of Late-Filed Second Amended Utah Contention Q and related exhibits were served on the persons listed below (unless otherwise noted) by e-mail with conforming copies by U.S. mail, first class, postage prepaid, this 3rd day of September 1999.

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Paul Gaukler
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AWMA

RADIOACTIVE WASTE MANAGEMENT ASSOCIATES

February 27, 1998

Charles Haughney, Director
Spent Fuel Project Office, Mail Stop 6F18
Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Holtec HI-STAR 100 TSAR
NRC Docket No. 72-1008

Dear Charley:

This letter concerns the g force that spent fuel cladding can withstand and the use of this parameter in safety analyses by Holtec, Sierra Nuclear and other cask manufacturers. This issue relates to the Holtec and Transtor storage/transport cask and transportation casks in general. In my opinion the most vulnerable fuel cannot withstand a 63g force in the most adverse orientation (Holtec TSAR, p. 3.5-1) but a force considerably less. At the very least, additional information should be requested from Holtec before issuing a Certificate of Compliance for the HI-STAR 100 cask. The Commission may also need to fund additional studies to consider this issue as it generally relates to transportation accidents involving irradiated fuel assemblies.

The "63 g" force for most vulnerable fuel is based on an analysis of the more ductile unirradiated, not irradiated, cladding. Despite the title of the Lawrence Livermore National Laboratory report on which Holtec relies ("Dynamic Impact Effects on Spent Fuel Assemblies," UCID-21246, October 1987), the LLNL report does not deal with "spent fuel" assemblies, only with non-irradiated fuel assemblies. As you are aware, irradiation within a reactor makes fuel assemblies more brittle and less resistant to impact. "Cladding ductility decreases and yield stress increases with increasing neutron fluence." ("Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Battelle Pacific Northwest Labs, NUREG/CR-5009, p. 2-5, February 1988).

LLNL's calculation for most vulnerable fuel also does not take into account the weight of the fuel itself, only the g force without the additional weight of the fuel. This considerable additional weight is an additional internal force. LLNL assumes fuel pellets remain in a rigid array in a high impact accident and will not impart a force to the cladding. This is obviously not correct.

NRC staff should ask Holtec and Sierra Nuclear to address this issue in their TSAR's. If no available studies analyze irradiated fuel cladding in high impact accidents, the NRC should fund additional studies to address this issue.

Marvin Resnikoff, Ph.D. ◆ Senior Associate
526 West 26th St. Rm 517 ◆ NY, NY 10001 ◆ 212 620-0526 ◆ Fax 212 620-0518 ◆ email radwaste@rwma.com

I wish these comments to be included in Holtec's NRC docket and to be considered in the Staff's safety evaluation report. Please send me a copy of the staff's draft safety evaluation report for the Holtec cask so that we may provide comments. If you have questions, feel free to call.

cc: D Curran
C Nakahara

Sincerely,

Marvin Resnikoff

72-1008 72-1014, 71-9261, 72-1023, 71-9268

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 19, 1998

Dr. Marvin Resnikoff, Senior Associate
Radioactive Waste Management Associates
526 West 26th Street, Room 517
New York, NY 10001

Dear Dr. Resnikoff:

I am responding to your February 27, 1998, letter regarding your concerns related to the structural integrity of spent fuel cladding under hypothetical accident conditions in spent fuel casks. In his March 11, 1998, letter, Charles J. Haughney, at the time, Acting Director, Spent Fuel Project Office, indicated the Nuclear Regulatory Commission (NRC) staff was reviewing your concerns and would report their findings to him to report directly to you. I apologize for the delay in responding to you; however, Mr. Haughney is currently serving in another office and several licensing actions took precedence in allocation of limited staff resources for completing the review. The staff has now completed its review of your concerns regarding the Lawrence Livermore National Laboratory (LLNL) Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," dated October 20, 1987, and determined that the LLNL report appeared to use sufficiently conservative data in the characterization of spent fuel cladding properties. The staff also found that the LLNL report conclusions appeared to be based on acceptable analysis and assumptions.

In particular, you stated that the LLNL report does not address irradiated fuel cladding, only unirradiated fuel cladding. In actuality, Table 3 of the report delineates irradiated cladding longitudinal tensile strength values. This table indicates that irradiated cladding has a greater strength value than unirradiated cladding. The LLNL report analysis used the values of unirradiated cladding strength, which is acceptable.

In your letter, you also stated that the LLNL report did not take into account the weight of the fuel assembly in the side drop orientation evaluation. In actuality, the fuel weight was delineated in Table 4 of the report and used appropriately in the analysis in Appendix A of the report. Thus, the LLNL report used the proper weight value in the analysis of the side drop orientation.

The NRC is committed to ensuring the safe operation of dry spent fuel storage and transport casks. The NRC staff will continue to evaluate industry data and analysis on spent fuel cladding properties in hypothetical accident conditions for these casks.

Please note that your letter has been placed in all applicable dockets (i.e., 72-1008, 72-1014, 71-9261, 72-1023, and 71-9268) and your questions and concerns will certainly be considered in the staff's safety evaluations of the pertinent cask designs. You will also have an opportunity to comment on the draft safety evaluation report for each cask design during the public comment period of federal rulemaking to incorporate that cask into Part 72 to Title 10 of the Code of Federal Regulations.

I trust this responds to your concerns. If you have additional questions or wish to discuss this matter further, please contact me at (301) 415-8518.

Sincerely,

ORIGINAL SIGNED BY /s/

Mark S. Delligatti, Senior Project Manager
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket Nos.: 72-1008, 72-1014, 71-9261,
72-1023, 71-9268

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AWMA

RADIOACTIVE WASTE MANAGEMENT ASSOCIATES

December 31, 1998

Mark S Delligatti, Senior Project Manager
Spent Fuel Licensing Section
NMSS
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mark:

Thank you for your November 19 response to my February 27 letter. Your letter did not fully answer my concerns, so I'll try once more.

Brittleness

From several NRC-contractor reports, it is my understanding that irradiated fuel cladding is more brittle than unirradiated fuel cladding. This should alter the consequences of a transportation or ISFSI accident involving impact. You stated that irradiated fuel cladding has "a greater strength value" than unirradiated fuel cladding, but this does not address my concerns about brittleness. It does not appear that NRC staff are querying Holtec and SNC about this important distinction between irradiated and unirradiated fuel cladding. Simply using unirradiated cladding strength in the Holtec and SNC SAR's may not be acceptable.

Dynamic Loading

I am aware that the fuel assembly weight is taken into account in the LLNL report and the Holtec SAR, but the loading is static, that is, the fuel weight is assumed to be evenly distributed along the cladding. The model is essentially a beam between two supports. But this model may not bound the physical situation. In a side impact, the cladding and the fuel are distinct beams. Under impact the fuel pellets would be expected to break their fixed configuration and strike the cladding with force. This dynamic loading is not considered in the LLNL report and may be important. It does not appear that NRC staff are querying Holtec and SNC about this important distinction between static and dynamic loading.

Thank you for reconsidering these issues. And best wishes for the new year.

Marvin Resnikoff

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STATE OF UTAH

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March 26, 1999

Secretary
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville MD 20852-2739

Sent via Federal Express &
E-mail: spd@nrc.gov

Attn: Rulemakings and Adjudications

re: Comments on Proposed Rule to add Holtec Hi-Star 100 Cask System to the List of Approved Spent Fuel Storage Casks.

Dear Secretary:

In response to 64 Fed. Reg. 1542, January 11, 1999, the State of Utah submits comments on the Preliminary Safety Evaluation Report and Proposed Certificate of Compliance for the Holtec HI-STAR 100 Storage Cask. These comments have been prepared with assistance from Marvin Resnikoff, Ph.D., Radioactive Waste Management Associates.

Sincerely,

Denise Chancellor
Assistant Attorney General

Enc:

Comments from the State of Utah

Preliminary Safety Evaluation Report and Proposed Certificate of Compliance

HI-STAR 100 Storage Cask

March 29, 1998

The State of Utah, with assistance from Marvin Resnikoff, Ph.D. of Radioactive Waste Management Associates, submits these comments on the preliminary Safety Evaluation Report (SER) and proposed Certificate of Compliance (CoC) for the Holtec HI-STAR 100 irradiated fuel storage cask, NRC Docket No. 72-1008. 64 Fed. Reg. 1542 (1999). The HI-STAR 100 is an all-metal cask with an outer metal overpack that encloses a sealed helium-filled canister (MPC) containing irradiated fuel. Although the SER and CoC under review here address storage only, the HI-STAR 100 is designed for both storage and transportation of spent nuclear power plant fuel.

The State has a three-fold interest in the adequacy of the SER and CoC for the HI STAR 100 storage cask. First, its design is virtually identical to the design of the HI STAR 100 transportation cask, which Private Fuel Storage L.L.C. (PFS) proposes to use to transport spent fuel to its proposed independent spent fuel storage installation (ISFSI) in Utah. The only difference between the storage cask and transportation cask is the fact that the transportation cask uses impact limiters and must satisfy hypothetical accident conditions under 10 CFR Part 71. Second, PFS plans to use the HI-STAR 100 storage cask's internal welded canister (multi-purpose canister or MPC) to transport and store fuel. The MPC will hold the irradiated fuel during transportation to the Private Fuel Storage facility. After arrival at the PFS facility, the MPCs will be stored in the HI-STORM 100 concrete overpack at the proposed PFS facility. Third, although PFS intends to use the HI-STORM-100 cask for storage under normal conditions, in case of an accident, the all-metal cask HI-STAR 100 cask will be used as a storage backup.

These comments address the conclusions of the SER, as well as the assertions made by the cask manufacturer, Holtec International, in the Technical Safety Analysis Reports (TSARs) for the HI-STAR and HI-STORM casks. The HI-STAR 100 storage TSAR is Holtec Report HI-941184 (NRC Docket No. 72-1008). The HI-STAR 100 transportation TSAR is Holtec Report HI-951251 (NRC Docket No. 71-9261). The HI-STORM 100 storage TSAR is Holtec Report HI-951312 (NRC Docket No. 72-1014).

The State is in the process of finalizing a confidentiality agreement with Holtec that will allow the State access to the Holtec proprietary version of HI -STAR 100 SAR, Revision 9 and HI-STORM 100 TSAR, Revision 5. The State will submit additional comments, as a proprietary and confidential submittal, after it has received and reviewed Holtec proprietary documents.

General Comments

The HI-STAR 100 design should not be approved, because Holtec has not provided reasonable assurance that the cladding and cask will retain their integrity under normal, off-normal and accident conditions. Moreover, Holtec does not correctly calculate health impacts under bounding accidents. Nor has Holtec evaluated the impact of a sabotage event. Finally, the TSAR and SER do not provide assurance the cask and cladding will retain their integrity under thermal conditions that exist at an ISFSI. Rather than addressing these deficiencies, the NRC's SER has glossed them over. These issues are crucially important to protecting the public health and safety, and therefore must be addressed before the Holtec CoC can be issued.

Specific Comments

Cladding Integrity Under Impact

According to the HI-STAR 100 storage TSAR (Sec. 3.5), the HI-STAR 100 system is designed to withstand a maximum deceleration of 60 g, while a Lawrence Livermore National Laboratories report shows that the most vulnerable fuel can withstand a deceleration of 63 g in the most adverse orientation (side drop).[1] Holtec therefore asserts that fuel rod integrity will be maintained under all accident conditions. In the preliminary SER (at 11-6), the NRC Staff concurs that "there is reasonable assurance that the cladding will maintain confinement integrity during a design basis drop."

In our view, this analysis is incorrect. Holtec and the NRC Staff have not demonstrated a reasonable assurance that the cladding will maintain its integrity.

Holtec's analysis does not provide reasonable assurance for the following reasons: (1) it does not take into account the possible increase in rate of oxidation of cladding of high burnup fuel; (2) Holtec relies for its analysis on a Lawrence Livermore National Laboratories (LLNL) report that fails to distinguish the effects of reactor irradiation on

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[1] UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies, Chum, Wm, Schwartz (October 20, 1987) (LLNL Report).

fuel assemblies; and (3) Holtec also relies on the LLNL Report's incorrect assumption that fuel assemblies act as a static rigid rod. The first factor (increased rate of oxidation) increases the likelihood that fuel cladding may rupture, and, when the three factors are taken together, they compound the likelihood of a release of radioactive materials during a foreseeable drop accident at the proposed ISFSI.

1. **Increased Rate of Oxidization.**

The NRC's Information Notice IN 98-29, entitled "Predicted Increase in Fuel Rod Cladding Oxidation" (August 3, 1998), provides new information, not considered in the Holtec TSAR, that calls into question Holtec's accident analysis. IN 98-29 discusses the oxidation rate of fuel cladding for high burnup fuel based on reported experiences with Westinghouse's fuel assemblies. NRC advises recipients to "review the information for applicability to their facilities and consider action as appropriate, to avoid similar problems." Nuclear Regulatory Commission, IN 98-29, *Predicted Increase in Fuel Rod Cladding Oxidation* (August 3, 1998) at 1.

The Notice reports that in October of 1997, Westinghouse notified NRC that modification of its fuel cladding corrosion model in its fuel design code to reflect new data on Zircaloy-4 oxidation at high burnup "may create compliance issues for its Integral Fuel Burnable Absorber (IFBA) fuel with Zircaloy-4 cladding." *Id.* at 1. As noted in the Information Notice,

> The modified code may predict higher fuel temperatures and internal pressures at high burnup conditions. This, in turn, may lead to code results that do not meet the Westinghouse criterion prohibiting gap reopening and that do not meet the loss-of-coolant accident (LOCA) criterion in 10 CFR 50.46(b)(2).

Id. Although the problem was initially discovered by Westinghouse with relation to Zircaloy 4 fuel, the Information Notice notes that "the burnup related phenomena, which could result in noncompliance with the oxidation requirements of 10 CFR 50.46, may not be limited to Westinghouse IFBA fuel but might affect any Zircaloy fuel used in high burnup application." IN 98-29 at 2. Thus, the experience at Westinghouse is also germane to any high burnup fuel that may be stored in Holtec casks--not just to Westinghouse fuel.

According to IN 98-92, the increased oxidation of the cladding is a function of the fuel burnup. Oxidization may cause the cladding to become effectively thinner, decreasing its structural integrity. This thinner cladding due to oxidization also lowers the 'g' impact force at which fuel cladding will shatter. Holtec's TSAR relies on the premise that fuel cladding will not shatter for any foreseeable drop. This premise is based on the assumption that it would take a side drop of greater than 63g to damage the cladding. Our spreadsheet calculations, presented below, show that the g loading for high burnup fuel with oxidized cladding approaches 45g. The NRC Staff should not approve the Holtec application unless and until Holtec has factored the information in IN 98-29 into its calculations. The clear implication of IN 98-29 is that the lift height of the HI-STAR 100 cask must be reduced to lower the g-forces on the cladding.

Effects of Changing Variables
Table 4 in *Dynamic Impact Effects on Spent Fuel Assemblies*

	A	B	C	D	E	F	G
Rod array	17x17	17x17	17x17	17x17	17x17	17x17	17x17
Assembly weight (lb)	1450.00	1450.00	1450.00	1450.00	1450.00	1450.00	1450.00
# of rods	264.00	264.00	264.00	264.00	264.00	264.00	264.00
Fueled length (in)	144.00	144.00	144.00	144.00	144.00	144.00	144.00
# of spacers (N)	7.00	7.00	7.00	7.00	7.00	7.00	7.00
L = (fueled length/(N-	24.00	24.00	24.00	24.00	24.00	24.00	24.00
E (psi)	1.04E+07	1.04E+07	1.04E+07	1.30E+07	1.04E+07	1.04E+07	1.04E+07
σy (psi)	8.05E+04	8.05E+04	8.05E+04	8.05E+04	4.50E+04	4.30E+04	8.05E+04
t (in)	0.02	0.02	0.02	0.02	0.02	0.02	0.05
ro (in)	0.19	0.19	0.18	0.19	0.19	0.19	0.21
ri (in)	0.16	0.16	0.16	0.16	0.16	0.16	0.16
A (in2)	0.02	0.02	0.02	0.02	0.02	0.02	0.05
I = (1/4*3.14(ro^4-	3.85E-04	3.85E-04	3.09E-04	3.85E-04	3.85E-04	3.85E-04	9.37E-04
W (lb)	0.84	0.84	0.69	0.84	0.84	0.84	1.78
w (lb/in)	0.04	0.04	0.04	0.04	0.04	0.04	0.04
r (in)	0.18	0.18	0.17	0.18	0.18	0.18	0.19
pressure (lb)	2250.00	1187.20	2250.00	2250.00	2250.00	1187.20	2250.00
σa (psi)	8787.50	4639.02	10472.14	8787.50	8787.50	4639.02	4675.00
M (lb-in)	2.32	2.32	2.32	2.32	2.32	2.32	2.32
σb (psi)	1128.70	1128.70	1378.17	1128.70	1128.70	1128.70	519.50
P (lb)	68.56	46.56	55.00	85.69	68.56	48.56	166.87
gL	81.93	81.93	80.06	102.41	81.93	81.93	93.71
gV	63.54	67.21	50.81	63.54	31.08	15.76	145.96

A: Values from Westinghouse specimen (Dynamic Impact Effects...Table 4)
B: Pressure changed to a lower value (value in An Assessment of the Risk...)
C: Thickness of fuel cladding decreased due to oxidation by 17%. Column A's thickness is reduced by 17%.
D: E Modulus changed to higher value (value in An Assessment of the Risk...)
E: Yield stress lowered to half the original value
F: Yield stress lowered and pressure lowered (E and B)
G: Doubling the thickness

Note:
DIE = Dynamic Impact Effects...
AAR = An Assessment of the Risk...
AAR's E modulus was expected to be lower, as they took irradiated zircaloy into account. However, it was not, after conversion.

	AAR	DIE
E modulus	1.30E+07	1.04E+07 psi
Pressure	1187.20	2250.00 psi

2. Irradiated and Unirradiated Fuel Assemblies.

Holtec's TSAR for the HI-STAR 100 storage cask relies for its *estimate of g force that* will damage fuel cladding upon a 1987 report by LLNL.[2] The LLNL Report fails to take into account the increased brittleness of irradiated fuel assemblies.[3] Because the irradiated fuel assemblies may have been embrittled, they would also be less resistant to impact. During the course of a fuel assembly's life, subatomic particle bombardment, including neutron flux, significantly decreases the assembly's ductility and increases the assembly's yield stress, thereby embrittling the fuel assembly. "Cladding ductility decreases and yield stress increases with increasing neutron fluence."[4]

Furthermore, the proposed HI-STAR 100 will store only irradiated fuel assemblies; thus, the Applicant cannot rely on LLNL's analysis because the LLNL does not account for irradiation and embrittlement, which lower the impact resistance of the fuel assemblies.

These facts are significant when coupled with the increased oxidation rate reported in IN 98-29 because increased oxidation could tangentially cause an increase in cladding embrittlement.[5] Thus, IN 98-29 compounds the LLNL's error in disregarding the brittle characteristics of irradiated fuel cladding.

3. Fuel Assemblies Do Not Act as a Rigid Rod.

Holtec's calculations rely upon LLNL's erroneous assumption that the fuel within the cladding behaves as a rigid rod. Thus, Holtec merely used a static calculation instead of taking into account the dynamic loading upon impact. The LLNL Report specifically states, "It is important to emphasize that the g loadings shown in Figs. 6 and 7 are static loadings."[6] This assumption is incorrect. Instead of a homogeneous, rigid rod, the fuel rod consists of fuel pellets stacked like coins within thin tubing. In any impact scenario, the fuel assembly does not act as a rigid rod; rather, it acts as a dynamic system with the fuel impacting the inside of the cladding and creating a greater likelihood of cladding rupture. Holtec has not shown that the assumption of a rigid rod is conservative. The thinner cladding due to the increased oxidation serves to compound this effect because a smaller g force would be required to rupture the assembly. The NRC staff should not approve the

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[2] LLNL Report.

[3] See e.g. UCID-21246, Table 4, which makes no distinction between Young's modulus and yield strength of a range of fuel assemblies.

[4] "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Battelle Pacific Northwest Lab, NUREG/CR-5009 (February 1988)

[5] Thin cladding acquires brittle characteristics at a faster rate than thicker cladding during fuel life. See IN 98-29 at 2 ("If this total oxidation limit were to be exceeded during an accident, the cladding could become embrittled.").

[6] LLNL Report.

Holtec application without a showing by the applicant that its calculations are conservative.

In sum, the newly discovered findings at Westinghouse, as recognized in the NRC's Notice, and the other concerns discussed above, raise significant questions about the adequacy of Holtec's accident analysis.

Health Impact of Accidents

The calculated health impacts under hypothetical accident conditions, discussed in Chapter 7 of Holtec's HI-STAR 100 TSAR, are not conservative. Three issues need to be more fully examined by NRC Staff: the design basis accident, the radiation pathways, and the dose to children.

1. Design basis accident.

Holtec's hypothetical design basis accident condition assumes 100% of the fuel rods are non-mechanically ruptured and the gases and particulates in the fuel rod gap between the cladding and fuel pellet are released to the MPC cavity and then to the external environment. Radiation doses are calculated 100 m from the cask. In the time interval between production of Rev. 4 and Rev. 6 of the TSAR, the NRC Staff requested Holtec to conduct the dose calculations in conformance with the final version of NUREG-1536.[7] The accident analysis in the final version of NUREG-1536 increased the amount of radioactivity to the MPC cavity by 5 orders of magnitude and would have placed doses at 100 m over the EPA's limit of 5 rem. In Rev. 6, Holtec responded to the NRC's request by changing the method of calculating doses to incorporate an extremely small cask leakage rate, rather than assuming 100% of the cask cavity was released to the external environment. TSAR, Rev. 6 at 11.2-15. Thus, Holtec's new analysis increased the radioactivity released to the cask cavity by 5 orders of magnitude, but the leakage rate reduced the amount released from the cask cavity to the environment by more than 5 orders of magnitude. The net effect of this sleight of hand, was to change the design basis accident, so as to reduce the doses to the thyroid and whole body at 100 m. In essence, the NRC staff has allowed the applicant to change the definition of a bounding accident to one that involves 100% fuel rod cladding rupture, with the cask lid intact, i.e., only slight leakage from the cask. The design basis accident no longer represents a loss-of-confinement-barrier accident.

Holtec's attempt to change the design basis accident for storage casks is not only inappropriate, but is unsupported. We strongly disagree that the slight cask leakage, 1.5 x

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[7] Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

10^{-4} cm³/s constitutes a bounding accident. A scenario that could lead to a greater release rate is a welding error that allows helium to leak from the MPC if a cask is dropped. Leakage of helium will allow the maximum cladding temperature to rise and the fuel rod cladding to rupture. In this case, the percentage of fuel rods that rupture may be less, but the leakage rate from the cask cavity would be greater than assumed by Holtec.

2. **Radiation pathways excluded**

In Chapter 7, Holtec has calculated the radiation dose to an adult 100 m from the accident, due solely to inhalation of the passing cloud. Other relevant pathways, such as direct radiation from cesium and cobalt-60 deposited on the ground, resuspension of deposited radionuclides, ingestion of contaminated food and water and incidental soil ingestion, are not considered, in violation of 10 CFR 72.24(m).

3. **Dose to children not considered**

Contrary to the standards in 10 C.F.R. Parts 72 and 20, Holtec has not calculated the dose to children. These standards prescribe dose limits for "an individual outside the controlled area" (10 C.F.R. § 72.24(m)), and "individual members of the public" (10 C.F.R. §§ 20.1301, 20.1302). For purposes of the Part 20 dose standards, the regulations define "individual" as "*any* human being," and "member of the public" as *any* individual except when that individual is receiving an occupational dose." (Emphasis added). The concept of "any individual" clearly includes people other than adult men, *i.e.*, children. Nor does the Atomic Energy Act limit its protection against undue risk to adult males. In fact, NRC regulations already make special exception for the dose to a minor (10 CFR § 20.1207) and the dose to an embryo/fetus (10 CFR § 20.1208) within restricted areas. Further, Regulatory Guide 3.51, "Calculational Models for Estimating Radiation Doses to Man from Airborne Radioactive Materials Resulting from Uranium Milling Operations," also calculates the dose to children and infants by adjusting the organ size, breathing rate and dose conversion factors.

Children are more vulnerable to radiation than adults because of their higher surface-area-to-volume of organs ratio.[8] Other contributing factors include the fact that children have higher soil ingestion rates than adults.[9] Children also have reduced ingestion and inhalation rates compared to adults;[10] nevertheless, the dose to children under a design basis accident is likely to be significantly higher than the dose to an adult. Thus, in order to satisfy the regulations and the Atomic Energy Act, it is necessary to determine whether

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[8] International Commission on Radiological Protection, "1990 Recommendations of the International Commission on Radiological Protection," ICRP 60, 1991, Pergamon Press.
[9] EPA, "Risk Assessment Guidance for Superfund: Volume I – Human Health Evaluation Manual (Part B, Development of Risk-based Preliminary Remediation Goals," EPA/540/R-92/003, December 1991.
[10] Eckerman, KF et al, "Health Risks from Low-Level Environmental Exposure to Radionuclides," Federal Guidance Report No. 13, Part I – Interim Version, prepared for the EPA, 1998.

the dose limits are satisfied for children. In addition, children are at a higher risk than adults of developing cancer because children live longer than adults and their cells grow more rapidly than adults' cells.

Note that it is not the regulation of 25 mr/yr or 100 mr/y or the EPA accident dose limits of 5 rems that are deficient. Rather, it is the NRC Staff's methodology in calculating exposures to children.

Sabotage Event

We disagree that an accident involving 100% fuel rod cladding rupture with slight lid leakage is a bounding accident. *See supra*, discussion on design basis accident. We urge NRC staff to consider the effect of a sabotage event with an anti-tank missile. The NRC already considers the impact of a tornado missile and explosion, but a tornado missile, like an 8" diameter steel rod striking the cask at 126 mph,[11] does not have the impact of an anti-tank missile. Similarly, an explosion with an external pressure of 300 psig does not have the impact of an anti-tank missile. The lack of a comprehensive assessment of the risks of sabotage and terrorism against nuclear waste facilities and shipments is well established. Terrorists have shown that they are capable of exploiting the weak interfaces associated with transportation as well as causing tremendous damage to static structures such as the World Trade Center and the Oklahoma City government building. As NRC Staff is aware, German regulatory authorities have imposed an additional condition on casks, namely, that they be able to withstand the impact of a 1-ton missile impacting a cask at the speed of sound. By this condition, German casks are able to withstand the impact of a jet engine striking a cask. NRC staff could impose additional conditions on dry storage casks and ISFSIs, *e.g.*, the CoC could require that an ISFSI be designed with an earthen berm to remove the line-of-sight.

Since the early 1980s, the NRC has relied on and has poorly interpreted an outdated set of experiments carried out by Sandia and Battelle Columbus Laboratories that measured the release of radioactive materials as a result of cask sabotage. In one of the Sandia experiments, a GE IF-300 truck cask containing one unirradiated fuel assembly was attacked with an M3A1, a military "shaped charge." Although the results "demonstrated that casks could indeed be breached by military explosives and that a considerable fraction of spent fuel could be released by such an attack,"[12] the NRC responded to Sandia's findings by concluding that since only 2/1,000,000 of the total fuel weight was released in inhalable form, the "average radiological consequences of a release in a heavily populated urban area such as New York City would be no early fatalities and less

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[11] Holtec HI-STAR 100 Storage TSAR, Table 2.2.5.

[12] Halstead, Robert J. and James David Ballard, "Nuclear Waste Transportation Security and Safety Issues: The Risk of Terrorism and Sabotage Against Repository Shipments," prepared for the Nevada Agency for Nuclear Projects, Carson City, Nevada, October, 1997, p.25.

that one (0.4) latent cancer fatality."[13] But this analysis is highly deficient, making a large number of questionable assumptions regarding evacuation and failing to include several significant radiation pathways, such as direct gamma exposures from deposited radionuclides. A more recent analysis[14] of a transportation accident in a rural area for a cask holding 14 (not 24) PWR fuel assemblies shows a cost of $620 million and a recovery time of 460 days for the cleanup operation. This cleanup is only to a level that would reduce doses to 500 mrem/year. Based on this accident scenario in a rural setting, a properly conducted, realistic accident scenario in an urban area can be expected to show billions of dollars in cleanup costs and lost revenues. An urban accident would also cause a large number of adverse health effects. Deficient as the analysis of a sabotage event during transportation is, the NRC Staff has never estimated the economic and safety implications of a sabotage event at a fixed storage facility.

The NRC has never explained why it considered the Sandia experiment indicative of what could occur in any type of terrorist attack, no matter the circumstances. Following the publication of these Sandia results, the NRC proposed elimination of many of the safety requirements for shipments of spent fuel aged more than 150 days, such as, no armed guards for the shipments in highly populated areas, no advance notice to the NRC or local law enforcement officials, and no periodic communication between escorts and a communications center.[15] "At least 32 parties submitted more than 100 pages of comments in response to the notice," to which the NRC never publicly responded. *See supra*, Halstead and Ballard, note 12.

In the intervening years since the Sandia experiments, anti-tank weapons with much greater accuracy and penetrating power have been manufactured and widely distributed. These devices could release much more radioactive material. The NRC suspended action on the rule-making, but it inappropriately continues to use the unrevised conclusions in the proposed rule as a basis for its policies on terrorism and sabotage of nuclear shipments.

The numerous terrorist attacks of the last several years have graphically demonstrated that the NRC continues to ignore the risks of sabotage at significant peril to the public. The NRC should adopt the specific recommendations of Halstead and Ballard for creating a realistic, up-to-date terrorism risk assessment. Some of the reference parameters Halstead and Ballard suggest are as follows:

- The reference weapon should be portable anti-tank missiles for their ability to permeate the strong cask materials, their range and availability.

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[13] *Id.*, at 26.

[14] Sandquist, GM et al. "Exposures and Health Effects from Spent Fuel Transportation," Rogers & Associates, RAE-8339/12-1, November 28, 1985.

[15] Halstead and Ballard at 27.

- A 10-year-cooled, medium burn-up, Westinghouse PWR assembly should be the reference spent fuel. A Holtec HI-STAR 100 storage cask loaded with 24 PWR assemblies of the reference fuel would represent a total radioactivity of about 5.5 million Curies. A terrorist incident resulting in a one-percent release would have radiological consequences far greater than those assumed in the HI-STAR TSAR.

- The following two scenarios, at a minimum should be considered: "an attack in which the cask is captured, penetrated by one or more explosive devices, and releases a significant amount (at least one percent) of its radioactive contents; and an attack in which the cask is perforated by one or more armor-piercing rockets or missiles and releases a significant amount (at least one percent) of its radioactive contents."[16]

Note that Halstead and Ballard recommend a 1% release because that is the percentage of unirradiated fuel released in the Sandia sabotage tests.[17] We maintain that a design basis accident should not be the release of 2×10^{-5} or less of the cesium inventory, but 1%, based on the Sandia sabotage tests. Further, it is not simply inhalable-sized particulates that are important. Larger-sized particulates will be released and deposited downwind, giving rise to a direct gamma dose.

Thermal Requirements

The proposed CoC temperature conditions for the Holtec HI-STAR 100 storage cask are not sufficient to guarantee that cladding and neutron shield degradation will be minimized. To reduce high temperatures, NRC staff must incorporate an additional condition into the CoC, a minimum pitch or center-to-center distance between casks. While Holtec has suggested a pitch of 12' or a 4' spacing between casks, this analysis is likely not based on rigorous calculations. Until the State receives the proprietary calculations from Holtec, it cannot comment with specificity on them. However, based on review of similar proprietary calculations for the HI-STORM 100 casks we have reviewed, we are skeptical that the proprietary calculations for the HI-STAR 100 cask are rigorous and sufficient.

Under the present regulatory framework, NRC staff and contractors must show that individual casks will not overheat if subjected to normal (average T = 80 °F) and off-normal (average T = 100 °F) temperatures. If the normal or off-normal temperature conditions are satisfied, then the cask may be used in that location. This is similar to the approach for the CoC earthquake and tornado conditions, but with one important difference: individual casks may interact with each other, causing temperature conditions above ambient temperature conditions. As a result, the Holite neutron absorbing material

[16] *Id.* at xix.
[17] Sandoval, RP *et al. An Assessment of the Safety of Spent Fuel Transportation in Urban Environs*, SAND82-2365, prepared for DOE by Sandia Labs, June 1983.

and the cladding may degrade due to excessive heat. In the HI-STAR 100 TSAR, the presence of adjacent casks and the concrete pad may not be correctly taken into account, as far as one can determine from Holtec's sketchy nonproprietary analysis. This should be properly addressed in the SER and CoC.

If the center-to-center distance between adjacent HI-STAR 100 casks is too small, casks may thermally interact with each other, effectively increasing the ambient temperature. According to Holtec's TSAR, the overpack shell outside surface temperatures are 229 °F and 249 °F under normal and off-normal temperature conditions.[18] In the most extreme example, if adjacent casks are in immediate contact, instead of the ambient temperature being 80 °F under normal conditions, it would be 229 °F. As the casks are moved away from each other, at some distance the casks become thermally independent of each other. Holtec attempts to calculate this distance in Fig. 4.4.5 by assuming a radiative blocking [illegible] due to the presence of other casks. But the situation at an ISFSI is far more complicated. It is not a blocking factor so much as the presence of adjacent heat sources at 229 °F. The effective ambient temperature will be raised as the casks interact with each other. The distance at which casks will act independently of each other must be calculated by Holtec and included in the CoC. For the HI-STAR 100 cask, the critical temperature is 300 °F for the inner surface of the Holite neutron absorbing material that surrounds the metal cask. The maximum temperatures of the Holite under normal and off-normal conditions are 274 °F and 294 °F, respectively. That is, the HI-STAR 100 is already operating with a thin safety margin, not accounting for the interaction between casks.

To take into account the interaction of casks, the following factors must be incorporated into the calculation. As a first approximation, Holtec could assume adjacent casks at the same temperature, $T_0 = 229$ °F. Insolation, average pad temperature, external convective air currents, and wind speed must also be incorporated into the model. The surface temperature of the center cask, T_1, could then be calculated. In the next iteration, the adjacent casks could be taken at temperature T_1 and a new temperature for the center cask could be calculated T_2. Holtec could then determine whether the series T_0, T_1, T_2 is converging to some asymptotic value. If the value for the inner surface of the neutron shield exceeds 300 °F, the casks must be spaced further apart.

As the situation presently stands, the SER and CoC are deficient. The maximum cladding temperature or temperature of the neutron shield inner surface has not be correctly calculated. NRC staff and Holtec are assuming there is no interaction between the casks. This assumption is not conservative.

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[18] Holtec, Topical Safety Analysis Report for the HI-STAR 100 Cask System, Holtec Report HI-941184, NRC Docket No. 72-1008, Table 11.1.

Removable Surface Contamination

The TSAR includes Technical Specifications for removable surface contamination.[19] If the smearable contamination exceeds 2200 dpm/100 cm^2 from gamma and beta emitting sources, the Technical Specifications require that the accessible surface be flushed or pressure washed. "If the smearable contamination limits still cannot be reduced to acceptable levels, evaluate and perform alternative actions up to, and including, removal of the MPC from the HI-STAR 100 overpack after removing the spent fuel from the MPC."[20] These conditions cannot be met at the proposed off-site Skull Valley ISFSI in Utah. No provisions exist for decontaminating casks under PFS's "start clean, stay clean" philosophy. PFS's proposed policy is to return casks that are contaminated above regulatory limits back to reactor sites. No provisions would exist at PFS for removing the MPC from the HI-STAR 100 overpack. In recognition of the conflict between the Tech Specs for the HI-STAR 100 and design of the PFS facility (and possibly other ISFSIs), the NRC should specify that all users of the HI-STAR 100 have the capability to remove smearable contamination onsite.

Future Rulemaking Procedures

The State of Utah strongly disagrees with any proposal by the NRC to approve future additions and revisions to the list of approved spent fuel storage casks as direct final rules. Under such a procedure there would be no proposed rule. Instead, the rule would become final within 75 days after publication unless NRC receives "significant adverse comments on the direct final rule within 30 days after publication." 64 Fed. Reg. at 1543. On receipt of such significantly adverse comments, NRC would withdraw the rule, address the comments, then publish a final rule. First, the premise underlying NRC's proposed procedural change -- that "additions and revisions to the list of approved spent fuel storage casks are noncontroversial and routine" -- is inaccurate. The above comments show that NRC's approval is not "routine." Moreover, given the past problems, such as hairline cracks associated with dry storage casks, it is imperative that future approval or revision to the list of approved casks be subject to adequate and rigorous public scrutiny. Second, a direct final rule reduces to 30 days the period of time for effective public comment. This is an insufficient time period to review and prepare comments that may be "significantly adverse" to cause NRC to withdraw the published final rule. Third, a direct final rule will diminish the public role in commenting and affecting the outcome of rulemaking procedure. It is more likely that NRC will give due consideration to comments at the propose rule stage; any comments at the final rule stage

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[19] *Id.* at 12.3-16.

[20] *Id.*

would have to be "significantly adverse" for NRC to reverse course and withdraw the direct final rule.

Safety considerations are too important for NRC to expedite the approval process at the expense of diminishing the public's role in commenting on the approval of spent nuclear fuel casks.

3.5 FUEL RODS

The cladding of the fuel rods is the initial confinement boundary in the HI-STORM 100 System. Analyses have been performed in Chapter 3 to ensure that the maximum temperature of the fuel cladding is below the Pacific Northwest Laboratory's threshold values for various cooling times. These temperature limits ensure that the fuel cladding will not degrade in an inert helium environment. Additional details on the fuel rod cladding temperature analyses for the spent fuel to be loaded into the HI-STORM 100 System are provided in Chapter 3.

The dimensions of the storage cell openings in the MPC are equal to or greater than those used in spent fuel racks supplied by Holtec International. Thousands of fuel assemblies have been shuffled in and out of these cells over the years without a single instance of cladding failure. The vast body of physical evidence from prior spent fuel handling operations provides confirmation that the fuel handling and loading operations with the HI-STORM 100 MPC will not endanger or compromise the integrity of the cladding or the structural integrity of the assembly.

The HI-STORM 100 System is designed and evaluated for a maximum deceleration of 45g's. Studies of the capability of spent fuel rods to resist impact loads [3.5.1] indicate that the most vulnerable fuel can withstand 63 g's in the side impact orientation. Therefore, limiting the HI-STORM 100 System to a maximum deceleration of 45 g's (perpendicular to the longitudinal axis of the overpack during all normal and hypothetical accident conditions) ensures that fuel rod cladding integrity is maintained. In [3.5.1], it is assumed that the fuel rod cladding provides the only structural resistance to bending and buckling of the rod. For accidents where the predominate deceleration is directed along the longitudinal axis of the overpack, [3.5.1] also demonstrates that no elastic instability or yielding of the cladding will occur until the deceleration level is well above the HI-STORM 100 limit of 45g's. The solutions presented in [3.5.1], however, assume that the fuel pellets are not intimately attached to the cladding when subjected to an axial deceleration load that may cause an elastic instability of the fuel rod cladding.

The limit based on classical Euler buckling analyses performed by Lawrence Livermore National Laboratory in [3.5.1] is 82 g's. In the LLNL report, the limiting axial load to ensure fuel rod stability is obtained by modeling the fuel rod as a simply supported beam with unsupported length equal to the grid strap spacing. The limit load under this condition is:

$$F = \pi^2 EI/L^2$$

In the preceding formula, E = Young's Modulus of the cladding, I = area moment of inertia of the cladding, and L = spacing of the grid straps.

Assuming that F = WxA/g with W being the weight of a fuel rod, and A = the deceleration, the Euler buckling formula can be expressed as

$$A/g = \pi^2 (ER^3tn/W_{fa}L^2) = \pi^2\beta$$

In the preceding formula, g = gravity, n = number of fuel rods in the fuel assembly, W_{fa} = the total weight of the fuel assembly, t = cladding wall thickness, and R = cladding mean radius.

Using the preceding formula, a survey of a large variety of fuel assembly types in [3.5.1] concluded that a 17 x 17 PWR assembly resulted in the minimum value for deceleration and results in the lower bound limit of:

$$A/g = 82$$

The fuel pellet weight was omitted from the analysis in [3.5.1] by virtue of the assumption that under axial load, the cladding did not support the fuel pellet mass. Since the results may not be conservative because of the assumption concerning the behavior of the fuel pellet mass, a new analysis of the structural response of the fuel cladding is presented here. It is demonstrated that the maximum axially oriented deceleration that can be applied to the fuel cladding is in excess of the design basis deceleration specified in this TSAR. Therefore, the initial confinement boundary remains intact during a hypothetical accident of transport where large axially directed decelerations are experienced by the HI-STORM 100 package.

The analysis reported in this section of the TSAR considers the most limiting fuel rod in the fuel assembly. Most limiting is defined as the fuel rod that may undergo the largest bending (lateral) deformations in the event of a loss of elastic stability. The fuel rod is modeled as a thin-walled elastic tube capable of undergoing large lateral displacements in the event that high axial loads cause a loss of stability (i.e., the non-linear interaction of axial and bending behavior of the elastic tube is included in the problem formulation). The fuel rod and the fuel pellet mass is included in the analysis with the fuel pellet mass assumed to contribute only its mass to the analysis. In the HI-STORM 100 spent fuel basket, continuous support to limit lateral movement is provided to the fuel assembly along its entire length. The extent of lateral movement of any fuel rod in a fuel assembly is limited to: (1) the clearance gap between the grid straps and the fuel basket cell wall at the grid strap locations; and, (2) the maximum available gap between the fuel basket cell wall and the fuel rod in the region between the grid straps. Note that the grid straps act as fuel rod spacers at the strap locations; away from the grid straps, however, there is no restraint against fuel rod –to-rod contact under a loading giving rise to large lateral motion of the individual rods. Under the incremental application of axial deceleration to the fuel rod, the fuel rod compresses and displaces from the axially oriented inertial loads experienced. The non-linear numerical analysis proceeds to track the behavior of the fuel rod up to and beyond contact with the rigid confining walls of the HI-STORM 100 fuel basket.

The analysis is carried out for the "most limiting" spent fuel assembly. The "most limiting" criteria used herein is based on the simple elastic stability formula assuming buckling occurs only between grid straps. This is identical to the methodology employed in [3.5.1] to identify the fuel assembly that limits design basis axial deceleration loading. Table 3.5.1 presents tabular data for a wide variety of fuel assemblies. Considerable data was obtained using the tables in [3.5.2]. The configuration with the lowest value of "Beta" is the most limiting for simple elastic Euler buckling between grid straps: the Westinghouse 14x14 Vantage,"W14V", PWR configuration is used to obtain results.

The material properties used in the non-linear analysis are those for irradiated Zircalloy and are obtained from [3.5.1]. The Young's Modulus and the cladding dynamic yield stress are set as:

$E = 10{,}400{,}000$ psi

$\sigma_y = 80{,}500$ psi

The fuel cladding material is assumed to have no tensile or compressive stress capacity beyond the material yield strength.

Calculations are performed for two limiting assumptions on the magnitude of resisting moment at the grid straps. Figures 3.5.1 through 3.5.9 aid in understanding the calculation. It is shown in the detailed calculations that the maximum stress in the fuel rod cladding occurs subsequent to the cladding deflecting and contacting the fuel basket cell wall. Two limiting analyses are carried out. The initial analysis assumes that the large deflection of the cladding between two grid straps occurs without any resisting moment at the grid strap supports. This maximizes the stress in the free span of the cladding, but eliminates all cladding stress at the grid strap supports. It is shown that this analysis provides a conservative lower bound on the limiting deceleration. The second analysis assumes a reasonable level of moment resistance to develop at the grid straps; the level developed is based on an assumed deflection shape for the cladding spans adjacent to the span subject to detailed analysis. For this second analysis, the limiting decelerations are much larger with the limit stress level occurring in the free span and at the grid strap support locations.

It is concluded that the most conservative set of assumptions on structural response still lead to the conclusion that the fuel rod cladding remains intact under the design basis deceleration levels set for the HI-STORM 100.

Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA

Array ID	Array Name	Rod O.D. (in.)	Clad Thk. (in.)	R_{mean} (in.)	# of Rods	Assy Wt. (lb.)	Rod Length (in.)	# of Spans	Average Span (in.)	Material Modulus	BETA
						PWR					
14x14A01	W14OFA	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14A02	W14OFA	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14A03	W14V	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14B01	W14STD	0.4220	0.0243	0.21708	179	1302	152.4	6	25.4	10400000	0.550863067
14x14B02	XX14TR	0.4170	0.0295	0.21588	179	1215	152	6	25.33333	10400000	0.708523868
14x14B03	XX14STD	0.4240	0.0300	0.21950	179	1271.2	149.1	8	18.6375	10400000	1.337586884
14x14C01	CE14	0.4400	0.0280	0.22700	176	1270	147	8	18.375	10400000	1.398051576
14x14C02	CE14	0.4400	0.0280	0.22700	176	1220	137	8	17.125	10400000	1.67556245
14x14D01	W14SS	0.4220	0.0165	0.21513	180	1247	126.68	6	21.11333	24700000	1.31385062
15x15A01	CE15P	0.4180	0.0260	0.21550	204	1360	140	9	15.55556	10400000	1.677523904
15x15B01	W15OFA	0.4220	0.0245	0.21713	204	1459	151.85	6	25.30833	10400000	0.569346561
15x15B02	W15V5H	0.4220	0.0245	0.21713	204	1459	151.85	6	25.30833	10400000	0.569346561
15x15B03	W15	0.4220	0.0243	0.21708	204	1440	151.83	6	25.305	10400000	0.571905185
15x15B04	W15	0.4220	0.0243	0.21708	204	1443	151.83	6	25.305	10400000	0.570716193
15x15B05	15(2a-319)	0.4220	0.0242	0.21705	204	1472	151.88	6	25.31333	10400000	0.556610964
15x15C01	SPC15	0.4240	0.0300	0.21950	204	1425	152	6	25.33333	10400000	0.73601861
15x15C02	SPC15	0.4240	0.0300	0.21950	204	1425	152	6	25.33333	10400000	0.73601861
15x15C03	XX15	0.4240	0.0300	0.21950	204	1432.8	152.065	6	25.34417	10400000	0.731386148
15x15C04	XX15	0.4170	0.0300	0.21600	204	1338.6	139.423	9	15.49144	10400000	1.996693327
15x15D01	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15D02	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15D03	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15G01	HN15SS	0.4220	0.0165	0.21513	204	1421	126.72	6	21.12	24700000	1.305875606
16x16A01	CE16	0.3820	0.0250	0.19725	236	1430	161	10	16.1	10400000	1.270423729

Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA (continued)

Array ID	Array Name	Rod O.D. (in.)	Clad Thk. (in.)	R_{mean} (in.)	# of Rods	Assy Wt. (lb)	Rod Length (in.)	# of Spans	Average Span (in.)	Material Modulus	BETA
16x16A02	CE16	0.3820	0.0250	0.19725	236	1300	146.499	9	16.27767	10400000	1.367126598
17x17A01	W17OFA	0.3600	0.0225	0.18563	264	1373	151.635	7	21.66214	10400000	0.613275783
17x17A02	W17OFA	0.3600	0.0225	0.18563	264	1365	152.3	7	21.75714	10400000	0.611494853
17x17B01	W17STD	0.3740	0.0225	0.19263	264	1482	151.635	7	21.66214	10400000	0.634902014
17x17B02	W17P+	0.3740	0.0225	0.19263	264	1482	151.635	7	21.66214	10400000	0.634902014
17x17C01	BW17	0.3790	0.0240	0.19550	264	1505	152.688	7	21.81257	10400000	0.687604262
					BWR						
6x6A02	XX/ANF6	0.5645	0.0360	0.29125	36	328.4	116.65	4	29.1625	10400000	1.192294364
6x6C01	HB6	0.5630	0.0320	0.28950	36	270	83	3	20.75	10400000	2.500527046
7x7A01	HB7	0.4860	0.0330	0.25125	49	276	83.2	3	20.8	10400000	2.233705011
7x7B01	GE-7	0.5630	0.0320	0.28950	49	682.5	159	7	19.875	10400000	1.467601583
7x7B02	GE-7	0.5630	0.0370	0.29075	49	681	164	7	20.5	10400000	1.619330439
7x7B03	GE-7	0.5630	0.0370	0.29075	49	674.4	164	7	20.5	10400000	1.635177979
7x7B04	GE-7	0.5700	0.0355	0.29388	49	600	161.1	7	20.1375	10400000	1.887049713
7x7B05	GE-7	0.5630	0.0340	0.29000	49	600	161.1	7	20.1375	10400000	1.736760659
8x8B03	GE-8	0.4930	0.0340	0.25500	63	681	164	7	20.5	10400000	1.2906798
8x8C02	GE-8R	0.4830	0.0320	0.24950	62	600	159	7	19.875	10400000	1.352138354
8x8C03	GE-8R	0.4830	0.0320	0.24950	62	600	163.71	7	20.46375	10400000	1.27545448
9x9D01	XX/ANF9	0.4240	0.0300	0.21950	79	575.3	163.84	8	18.20444	10400000	1.367212516
10x10E01	XX10SS	0.3940	0.0220	0.20250	96	376.6	89.98	4	17.996	24700000	3.551678654

Array ID, Rod OD, Clad Thk and # of Rods from Tables 6.2.1 and 6.2.2.
Rmean, Average Span and THETA are Calculated.
Zircaloy Modulus from LLNL Report [2.9.1].
Stainless Steel (348H) Modulus from ASME Code, Section III, Part D.

Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA (continued)

PWR Assy. Wt., Rod Len. and # of Spans (exc. as noted below) from DOE/RW-0184, Vol. 3, UC-70, -71 and -85, Dec. 1987

Assy. Wt., Rod Len. and # of Spans for 15x15B03, 15x15B04, 15x15C01 and 15x15C02 from ORNL/TM-9591/V1-R1.

BWR Assy. Wt., Rod Len. and # of Spans (exc. as noted below) from ORNL/TM-10902.

Assy. Wt., Rod Len. and # of Spans for 6x6A02, 9x9D01 and 10x10E01 from DOE/RW-0184, Vol. 3, UC-70, -71 and -85, Dec. 1987

Assy. Wt., Rod Len. and # of Spans for 7x7B04 and 7x7B05 from ORNL/TM-9591/V1-R1.

Assy. Wt. for 8x8C02 and 8x8C03 from ORNL/TM-9591/V1-R1

In the following, a physical description of the structural instability problem is provided with the aid of Figures 3.5.1 to 3.5.9. A stored fuel assembly consists of a square grid of fuel rods. Each fuel rod consists of a thin-walled cylinder surrounding and containing the fuel pellets. The majority of the total weight of a fuel rod is in the fuel pellets; however, the entire structural resistance of the fuel rod to lateral and longitudinal loads is provided by the cladding. Hereinafter, the use of the words "fuel rod", "fuel rod cladding", or just "cladding" means the structural thin cylinder. The weight of the fuel pellets is conservatively assumed to be attached to the cladding for all discussions and evaluations.

Figure 3.5.1 shows a typical fuel rod in a fuel assembly. Also shown in Figure 3.5.1 are the grid straps and the surrounding walls of the spent fuel basket cell walls. The grid straps serve to maintain the fuel rods in a square array at a certain number of locations along the length of the fuel assembly. When the fuel rod is subject to a loading causing a lateral deformation, the grid strap locations are the first locations along the length of the rod where contact with the fuel basket cell walls occurs. The fuel basket cell walls are assumed to be rigid surfaces. The fuel rod is assumed subject to some axial load and most likely has some slight initially deformed shape. For the purposes of the analysis, it is assumed that displacement under load occurs in a 2-D plane and that the ends of the fuel rod cladding have a specified boundary condition to restrain lateral deflection. The ends of the fuel rod cladding are assumed to be simply supported and the grid straps along the length of the fuel assembly are assumed to have gap "g_1" relative to the cell walls of the fuel basket. The figure shows a typical fuel rod in the assembly that is located by gaps "g_2" and "g_3" with respect to the fuel basket walls. Because the individual fuel rod is long and slender and is not perfectly straight, it will deform under a small axial load into the position shown in Figure 3.5.2. The actual axial load is due to the distributed weight subject to a deceleration from a hypothetical accident of transport. For the purposes of this discussion, it is assumed that some equivalent axial load is applied to one end of the fuel rod cladding. Because of the distributed weight and the fact that a deceleration load is not likely to be exactly axially oriented, the predominately axial load will induce a lateral displacement of the fuel rod cladding between the two end supports. The displacement will not be symmetric but will be larger toward the end of the cladding where support against the axial deceleration is provided. Depending on the number of grid straps, either one or two grid straps will initially make contact with the fuel basket cell wall and the contact will not be exactly centered along the length of the cell. Figure 3.5.3 illustrates the position of the fuel rod after the axial load has increased beyond the value when initial contact occurred and additional grid straps are now in contact with the cell wall. The maximum stress in the fuel rod will occur at the location of maximum curvature and will be a function of the bending moment (F_2 x(g_2-g_1)).

At some load $F_3 > F_2$, either the limit stress in the fuel rod cladding is achieved or the rod begins to experience large lateral movements between grid plates because of the coupling between axial and lateral load and deformation. Figure 3.5.4 shows the deformation mode experienced by the fuel rod cladding caused by the onset of an instability between two grid straps that are in contact with the fuel basket cell wall.

Once the lateral displacement initiates, the rod displaces until contact with the cell wall occurs at the mid point "A" (see Figure 3.5.5) or the cladding stress exceeds the cladding material yield strength. Depending on the particular location of the fuel rod in the fuel assembly, the highest stressed portion of the fuel rod will occur in the segment with the larger of the two gaps "g_2" and "g_3". For the discussion to follow, assume that $g_2 > g_3$. The boundary condition at the grid strap is conservatively assumed as simply-supported so that the analysis need not consider what happens in adjacent spans between grid straps. At this point in the loading process, the maximum bending moment occurs at the contact point and has the value F_4 x (g_2-g_1). Figure 3.5.5 shows the displaced configuration at the load level where initial contact occurs with the fuel cell wall. If the maximum fuel rod stress (from the bending moment and from the axial load) equals the yield stress of the fuel rod cladding, it is assumed that $F_3 = F_4$ is the maximum axial load that can be supported. The maximum stress in the fuel rod cladding occurs at point "A" in Figure 3.5.5 since that location has the maximum bending moment. If the cladding stress is still below yield, additional load can be supported. As the load is further increased, the bending moment is decreased and replaced by reaction loads, "V", at the grid strap and the contact point. These reaction loads V are shown in Figure 3.5.7 and are normal to the cell wall surface. Figure 3.5.6 shows the configuration after the load has been further increased from the value at initial contact. There are two distinct regions that need to be considered subsequent to initial contact with the fuel basket cell wall. During the additional loading phase, the point "A" becomes two "traveling" points, A, and A'. Since the bending moment at A' and A is zero, the moment F_5 x (g_2-g_1) is balanced by forces V at the grid strap and at point A or A'. This is shown in Figure 3.5.7 where the unsupported length current "a" is shown with the balancing load. At this point in the process, two "failure" modes are possible for the fuel rod cladding.

The axial load that develops in the unsupported region between the grid strap and point A' causes increased deformation and stress in that segment, or,

The straight region of the rod, between A and A', begins to experience a lateral deformation away from the cell wall.

Note that in this latter scenario, the slope at A or A' remains zero so this should never govern unless the flat region becomes large. The final limit load occurs when the maximum stress in either portion of the rod exceeds the yield stress of the tube. In what follows, the most limiting fuel assembly from the array of fuel types considered is subject to detailed analysis and the limit load established. This limit axial load is considered as the product of the fuel rod weight times the deceleration. Therefore, establishing the limit load to reach cladding material yield establishes the limiting axial deceleration that can be imposed.

The preceding discussion has assumed end conditions of simple support for conservatism. The location of the fuel rod determines the actual free gap between grid straps. For example, a fuel rod furthest from the cell wall that resists lateral movement of the assembly moves to close up all of the clearances that exist between it and the resisting cell wall. The clearance between rods is the rod pitch minus the rod diameter. In a 14 x 14 assembly, there are 13 clearance gaps plus an additional clearance g_3 between the nearest rod and the cell wall. Therefore, the gap g_2 is given as

$$g_2 = 13(\text{pitch-diameter}) + g_3$$

Figure 3.5.9 provides an illustration of the fuel rod deformation for a case of 5 fuel rods in a column. Clearly for this case, the available lateral movement can be considerable for the "furthest" fuel rod. On the other hand, for this fuel rod, there will be considerable moment resistance at the grid strap from the adjacent section of the fuel rod. The situation is different when the rod being analyzed is assumed to be the closest to the cell wall. In this case, the clearance gap is much smaller, but the moment resistance provided by adjacent sections of the rod is reduced. For calculation purposes, we assume that a moment resistance is provided as $M = f \times K\theta$ for the fuel rod under analysis where

$K = 3EI/L$, L= span between grid straps, and "f" is an assumed fraction of K

The preceding result for the rotational spring constant assumes a simple support at each end of the span with an end moment "M" applied. Classical strength of materials gives the result for the spring constant. The arbitrary assumption of a constant reduction in the spring constant is to account for undetermined interactions between axial force in the rod and the calculated spring constant. As the compressive force in the adjacent members increases, the spring constant will be reduced. On the other hand, as the adjacent span contacts its near cell wall, the spring constant increases. On balance, it should be conservative to assume a considerable reduction in the spring constant available to the span being analyzed in detail. As a further conservatism, we also use the angle θ defined by the geometry and not include any additional elastic displacement shape. This will further reduce the value of the resisting moment at any stage of the solution. In the detailed calculations, two limiting cases are examined. To limit the analysis to a single rod, it is assumed that after "stack-up" of the rods (see Figure 3.5.9), the lateral support provided by the cell wall supports all of the rods. That is, the rods are considered to have non-deforming cross-section.

Numerical Analysis - Based on the tabular results in Table 3.5.1, the fuel assembly with the smallest value for the deceleration based on the classical Euler buckling formula is analyzed in detail. The following input data is specified for the limiting 14 x 14 assembly [3.5.2]:

Inside dimension of a HI-STORM 100 fuel basket cell $s := 8.75 \cdot in$

Outside envelope dimension of grid plate $gp := 7.763 \cdot in$

Outer diameter of fuel rod cladding $D := .4 \cdot in$

Wall thickness of cladding $t := .0243 \cdot in$

Weight of fuel assembly(including end fittings) $W := 1177 \cdot lbf$

Number of fuel rods + guide/instrument tubes in a column or row $n := 14$

Overall length of fuel rod between assumed end support $L_t := 151 \cdot in$

Length of fuel rod between grid straps $L_s := 25.3 \cdot in$

Average clearance to cell wall at a grid strap location assuming a straight and centered fuel assembly $g_1 := .5 \cdot (s - gp)$

$g_1 = 0.494 \cdot in$

Rod pitch $pitch := 0.556 \cdot in$

$Clearance := (n - 1) \cdot (pitch - D)$ $Clearance = 2.028 \cdot in$

Minimum available clearance for lateral movement of a fuel rod between grid straps

$$g_3 := g_1 + .5 \cdot (gp - (n \cdot D + Clearance))$$

$g_3 = 0.561 \cdot in$

Maximum available clearances for lateral movement of a fuel rod between grid straps $g_2 := g_3 + Clearance$

$g_2 = 2.589 \cdot in$

Young's Modulus of Zircalloy [3.5.1] $E := 10400000 \cdot psi$

Dynamic Yield Strength of Zircalloy [3.5.1] $\sigma_y := 80500 \cdot psi$

Geometry Calculations:

Compute the metal cross section area A, the metal area moment of inertia I, and the total weight of a single fuel rod (conservatively assume that end fittings are only supported by fuel rods in the loading scenario of interest).

$$A := \frac{\pi}{4} \cdot \left[D^2 - (D - 2 \cdot t)^2\right] \qquad I := \frac{\pi}{64} \cdot \left[D^4 - (D - 2 \cdot t)^4\right]$$

$$A = 0.029 \cdot in^2 \qquad I = 5.082 \cdot 10^{-4} \cdot in^4$$

$$W_r := \frac{W}{n^2} \qquad W_r = 6.005 \cdot lbf$$

As an initial lower bound calculation, assume no rotational support from adjacent spans and define a multiplying factor

$$f := 0.0$$

Compute the rotational spring constant available from adjacent sections of the rod.

$$K := 3 \cdot E \cdot \frac{I}{L_s} \cdot f \qquad K = 0 \cdot lbf \cdot in$$

Now compute the limit load, if applied at one end of the fuel rod cladding, that causes an overall elastic instability and contact with the cell wall. Assume buckling in a symmetric mode for a conservatively low result. The purpose of this calculation is solely to demonstrate the flexibility of the single fuel rod. No resisting moment capacity is assumed to be present at the fittings.

$$P_0 := \pi^2 \cdot E \cdot \frac{I}{L_t^2} \qquad P_0 = 2.288 \cdot lbf$$

Note that this is less than the weight of the rod itself. This demonstrates that in the absence of any additional axial support, the fuel rod will bow and be supported by the cell walls under a very small axial load. In reality, however, there is additional axial support that would increase this initial buckling load. The stress induced in the rod by this overall deflected shape is small.

$$Stress_1 := \frac{P_0 \cdot g_1 \cdot D}{2 \cdot I} \qquad Stress_1 = 444.32 \cdot psi$$

$$Stress_d := \frac{P_0}{A} \qquad Stress_d = 79.76 \cdot psi$$

The conclusion of this initial calculation is that grid straps come in contact and we need only consider what happens between a grid strap. We first calculate the classical Euler buckling load based on a pin-ended rod and assuming conservatively that the entire weight of the rod is providing the axial driving force. This gives a conservatively low estimate of the limiting deceleration that can be resisted before a perfectly straight rod buckles.

$$a_{lim1} := \pi^2 \cdot E \cdot \frac{I}{L_s^2 \cdot W_r} \qquad a_{lim1} = 13.57$$

The rigid body angle of rotation at the grid strap under this load that causes contact is:

$$\theta_1 := \operatorname{atan}\left[2 \cdot \frac{(g_2 - g_1)}{L_s}\right] \qquad \theta_1 = 9.406 \cdot deg$$

Conservatively assume resisting moment at the grid is proportional to this "rigid body" angle:

$M_r := K \cdot \theta_1$ $\qquad M_r = 0 \cdot in \cdot lbf$ $\qquad$ (in this first analysis, no resisting moment is assumed)

The total stress at the grid strap due to the axial force and the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim1}}{A} + \frac{M_r \cdot D}{2 \cdot I} \qquad \sigma_{gs} = 2841.172 \cdot psi$$

The total stress at the contact location is

$$\mathrm{Stress}_2 := \frac{\left[W_r \cdot a_{lim1} \cdot (g_2 - g_1) - M_r\right] \cdot D}{2 \cdot I} \qquad \mathrm{Stress}_2 = 6.721 \cdot 10^4 \cdot \mathrm{psi}$$

$$\mathrm{Stress}_{2d} := \frac{W_r \cdot a_{lim1}}{A} \qquad \mathrm{Stress}_{2d} = 2841.172 \cdot \mathrm{psi}$$

$$\mathrm{Stress}_{2t} := \mathrm{Stress}_2 + \mathrm{Stress}_{2d} \qquad \mathrm{Stress}_{2t} = 7.005 \cdot 10^4 \cdot \mathrm{psi}$$

This is the maximum value of the stress at this location since, for further increase in axial load, the moment will decrease with consequent large decrease in the total stress.

The safety factor is

$$\frac{\sigma_y}{\mathrm{Stress}_{2t}} = 1.149$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{\left(W_r \cdot a_{lim1}\right)}{\cos\left(\theta_1\right)} \qquad P_{ax} = 82.599 \cdot \mathrm{lbf}$$

At this point in the load process, a certain axial load exists in the unsupported span on either side of the contact point. However, since the unsupported span is approximately 50% of the original span, the allowable deceleration limit is larger. As the axial load is incrementally increased, the moment at the contact point is reduced to zero with consequent increases in the lateral force V at the grid strap and at the contact points A and A'. Figure 3.5.8 provides the necessary information to determine the elastic deformation that occurs in the unsupported span as the axial load increases and the contact points separate (and, therefore, decreasing the free span).

From geometry, coupled with the assumption that the deflected shape is a half "sin" function with peak value "δ", the following relations are developed:

Assume "a" is a fraction of 50% of the span (the following calculations show only the final iterated assumption for the fraction

$$\varepsilon := .9 \qquad a := \varepsilon \cdot \left(\frac{L_s}{2}\right) \qquad a = 11.385 \cdot \mathrm{in}$$

Calculate "b" in Figure 3.5.8

$$b := \left[(a)^2 + (g_2 - g_1)^2\right]^{.5} \qquad b = 11.576 \cdot in$$

an equation for δ can be developed from the geometric relation

$$\frac{(g_2 - g_1)}{a} := \frac{b}{2\,(R - \delta)}$$

The inverse of the radius of curvature, R, at the point of peak elastic deflection of the free span, is computed as the second derivative of the assumed sin wave deflection shape. Based on the geometry in Figure 3.5.8, the peak deflection is:

$$\delta := .5 \cdot \left[\left[a \cdot \frac{b}{2 \cdot (g_2 - g_1)}\right]^2 + 4 \cdot \left(\frac{b}{\pi}\right)^2\right]^{.5} - a \cdot \frac{b}{4 \cdot (g_2 - g_1)}$$

$$\delta = 0.426 \cdot in$$

For the assumed "a", the limiting axial load capacity in the unsupported region is conservatively estimated as:

$$a_{lim2} := \pi^2 \cdot E \cdot \frac{I}{(b)^2 \cdot W_r} \qquad a_{lim2} = 64.816$$

The corresponding rigid body angle is:

$$\theta_2 := \text{atan}\left[1 \cdot \frac{(g_2 - g_1)}{a}\right] \qquad \theta_2 = 10.429 \cdot deg$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{\left(W_r \cdot a_{lim2}\right)}{\cos\left(\theta_2\right)} \qquad P_{ax} = 395.763 \cdot lbf$$

The resisting moment is

$$M_r := K \cdot \theta_2 \qquad M_r = 0 \cdot in \cdot lbf$$

The total stress in the middle of the unsupported section of free span "b" is

$$stress_3 := \frac{\left(P_{ax} \cdot \delta - M_r\right) \cdot D}{2 \cdot I} \qquad stress_3 = 6.635 \cdot 10^4 \cdot psi$$

$$stress_{3d} := \frac{P_{ax}}{A} \qquad stress_{3d} = 1.38 \cdot 10^4 \cdot psi$$

$$stress_{3t} := stress_3 + stress_{3d} \qquad stress_{3t} = 8.015 \cdot 10^4 \cdot psi$$

The safety factor is $\dfrac{\sigma_y}{stress_{3t}} = 1.004$

The total stress at the grid strap due to the axial force and any the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim2}}{A} + \frac{M_r \cdot D}{2 \cdot I} \qquad \sigma_{gs} = 1.357 \cdot 10^4 \cdot psi$$

The safety factor is $\dfrac{\sigma_y}{\sigma_{gs}} = 5.932$

For this set of assumptions, the stress capacity of the rod cladding has been achieved, so that the limit deceleration is:

$$A_{limit} := a_{lim2} \qquad A_{limit} = 64.816$$

This exceeds the design basis for the HI-STORM 100 package.

If there is any restraining moment from the adjacent span, there is a possibility of exceeding the rod structural limits at that location due to the induced stress. Therefore, the above calculations are repeated for an assumed moment capacity at the grid strap.

$f := 1.$ $\qquad K := 3\cdot E\cdot\frac{I}{L_s}\cdot f$

The rigid body angle of rotation at the grid strap under this load that causes contact is:

$$\theta_1 := \mathrm{atan}\left[2\cdot\frac{(g_2 - g_1)}{L_s}\right] \qquad \theta_1 = 9.406\,\mathrm{deg}$$

Conservatively assume resisting moment at the grid a function of this angle,is

$$M_r := K\cdot\theta_1 \qquad M_r = 102.875\,\mathrm{in\cdot lbf}$$

The total stress at the grid strap due to the axial force and the resisting moment is

$$\sigma_{gs} := \frac{W_r\cdot a_{lim1}}{A} + \frac{M_r\cdot D}{2\cdot I} \qquad \sigma_{gs} = 4.333\cdot 10^4\,\mathrm{psi}$$

The total stress at the contact location is

$$Stress_2 := \frac{\left[W_r\cdot a_{lim1}\cdot(g_2 - g_1) - M_r\right]\cdot D}{2\cdot I} \qquad Stress_2 = 2.672\cdot 10^4\,\mathrm{psi}$$

$$Stress_{2d} := \frac{W_r\cdot a_{lim1}}{A} \qquad Stress_{2d} = 2841.172\,\mathrm{psi}$$

$$Stress_{2t} := Stress_2 + Stress_{2d} \qquad Stress_{2t} = 2.956\cdot 10^4\,\mathrm{psi}$$

This is the maximum value of the stress at this location since, for further increase in axial load, the moment will decrease with consequent large decrease in the total stress.

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{(W_r\cdot a_{lim1})}{\cos(\theta_1)} \qquad P_{ax} = 82.599\,\mathrm{lbf}$$

At this point in the load process, a certain axial load exists in the unsupported span on either side of the contact point. However, since the unsupported span is approximately 50% of the original span, the allowable deceleration limit is larger. As the axial load is incrementally increased, the moment at the contact point is reduced to zero with consequent increases in the lateral force V at the grid strap and at the contact points A and A'. Figure 3.5.8 provides the necessary information to determine the elastic deformation that occurs in the unsupported span as the axial load increases and the contact points separate (and, therefore, decreasing the free span).

From geometry, coupled with the assumption that the deflected shape is a half "sin" function with peak value "δ", the following relations are developed:

Assume "a" is a fraction of 50% of the span (the following calculations show only the fina iterated assumption for the fraction

$\varepsilon := .7$ $\qquad a := \varepsilon \cdot \left(\frac{L_s}{2}\right)$ $\qquad a = 8.855 \cdot in$

Calculate "b" in Figure 3.5.8

$$b := \left[(a)^2 + \left(g_2 - g_1\right)^2\right]^{.5} \qquad b = 9.1 \cdot in$$

The inverse of the radius of curvature, R, at the point of peak elastic deflection of the free span, is computed as the second derivative of the assumed sin wave deflection shape. Based on the geometry in Figure 3.5.8, the peak deflection is:

$$\delta := .5 \cdot \left[\left[a \cdot \frac{b}{2 \cdot \left(g_2 - g_1\right)}\right]^2 + 4 \cdot \left(\frac{b}{\pi}\right)^2\right]^{.5} - a \cdot \frac{b}{4 \cdot \left(g_2 - g_1\right)}$$

$$\delta = 0.427 \cdot in$$

For the assumed "a", the limiting axial load capacity in the unsupported region is conservatively estimated as:

$$a_{lim2} := \pi^2 \cdot E \cdot \frac{I}{(b)^2 \cdot W_r} \qquad a_{lim2} = 104.9$$

The corresponding rigid body angle is:

$$\theta_2 := \text{atan}\left[1 \cdot \frac{(g_2 - g_1)}{a}\right] \qquad \theta_2 = 13.314 \cdot \text{deg}$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{(W_r \cdot a_{lim2})}{\cos(\theta_2)} \qquad P_{ax} = 647.331 \cdot \text{lbf}$$

The resisting moment is

$$M_r := K \cdot \theta_2 \qquad M_r = 145.619 \cdot \text{in} \cdot \text{lbf}$$

The total stress in the middle of the unsupported section of free span "b" is

$$\text{stress}_3 := \frac{(P_{ax} \cdot \delta - M_r) \cdot D}{2 \cdot I} \qquad \text{stress}_3 = 5.145 \cdot 10^4 \cdot \text{psi}$$

$$\text{stress}_{3d} := \frac{P_{ax}}{A} \qquad \text{stress}_{3d} = 2.257 \cdot 10^4 \cdot \text{psi}$$

$$\text{stress}_{3t} := \text{stress}_3 + \text{stress}_{3d} \qquad \text{stress}_{3t} = 7.402 \cdot 10^4 \cdot \text{psi}$$

The safety factor is $\dfrac{\sigma_y}{\text{stress}_{3t}} = 1.088$

The total stress at the grid strap due to the axial force and any the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim2}}{A} + \frac{M_r \cdot D}{2 \cdot I} \qquad \sigma_{gs} = 7.928 \cdot 10^4 \cdot psi$$

The safety factor is $\dfrac{\sigma_y}{\sigma_{gs}} = 1.015$

For this set of assumptions, the stress capacity of the rod cladding has been achieved, so that the limit deceleration is:

$$A_{limit} := a_{lim2} \qquad A_{limit} = 104.9$$

Conclusions

An analysis has demonstrated that for the most limiting PWR fuel assembly stored in the HI-STORM 100 fuel basket, a conservative lower bound limit on acceptable axial decelerations exceeds the 45g design basis of the cask. For a reasonable assumption of moment resisting capacity at the grid straps, the axial deceleration limit exceeds the design basis by a large margin.

It is concluded that fuel rod integrity is maintained in the event of a hypothetical accident condition leading to a 45g design basis deceleration in the direction normal to the target.

FUEL ROD DEFORMATION PHASES

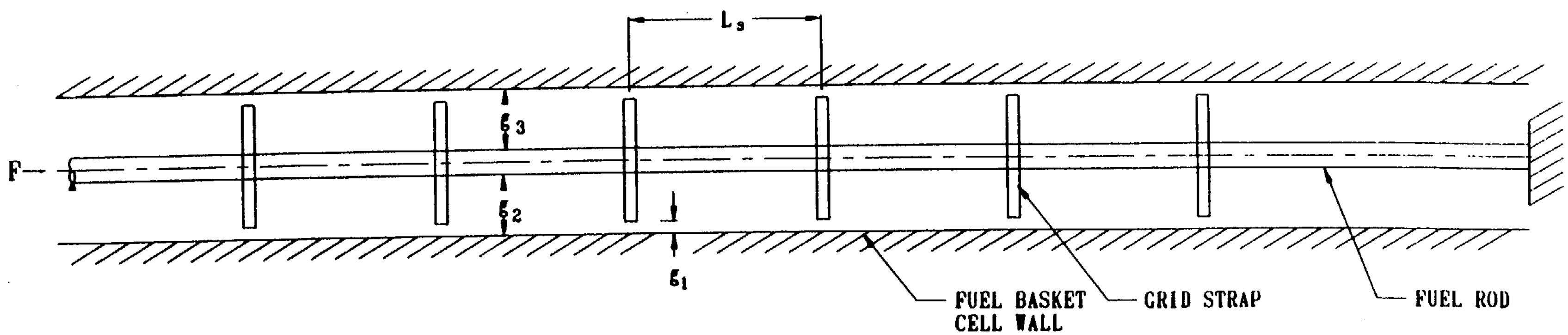


FIGURE 3.5.1; $g_1 > 0$

FUEL ROD DEFORMATION PHASES

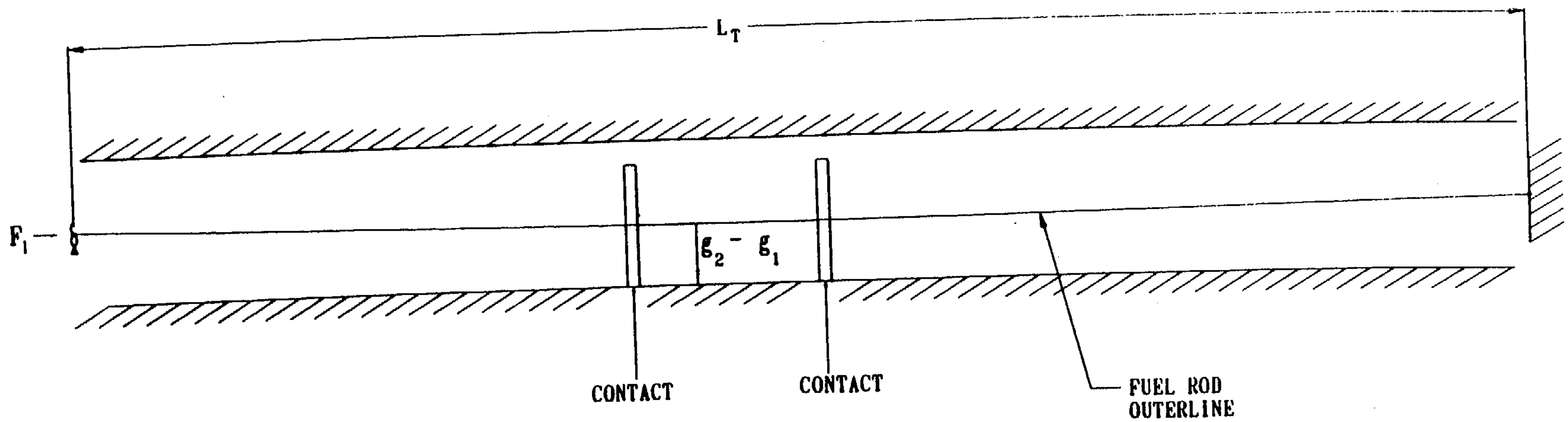


FIGURE 3.5.2; $g_1 = 0$

FUEL ROD DEFORMATION PHASES

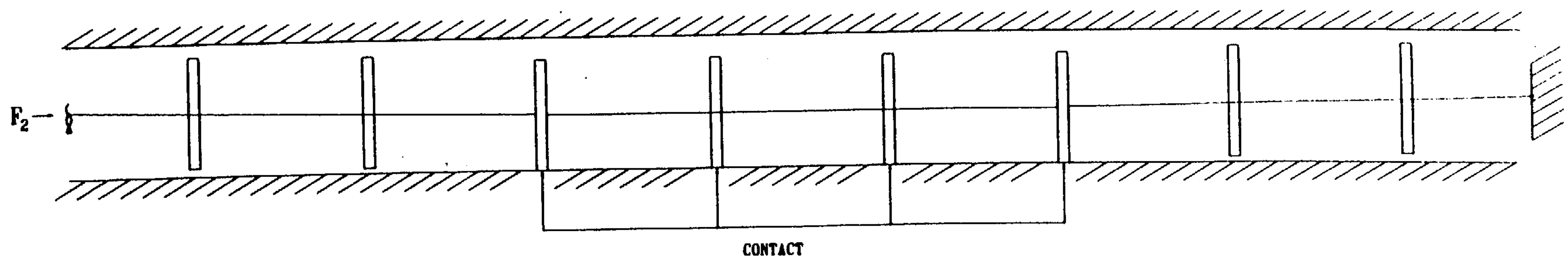


FIGURE 3.5.3; $g_1 = 0$, $F_2 > F_1$

FUEL ROD DEFORMATION PHASES

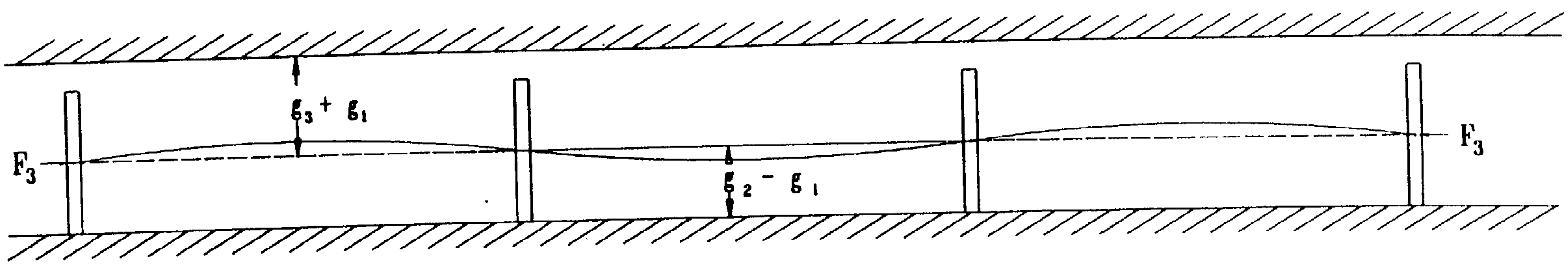


FIGURE 3.5.4; INTER-GRID STRAP DEFORMATION $F_3 > F_2$

FUEL ROD DEFORMATION PHASES

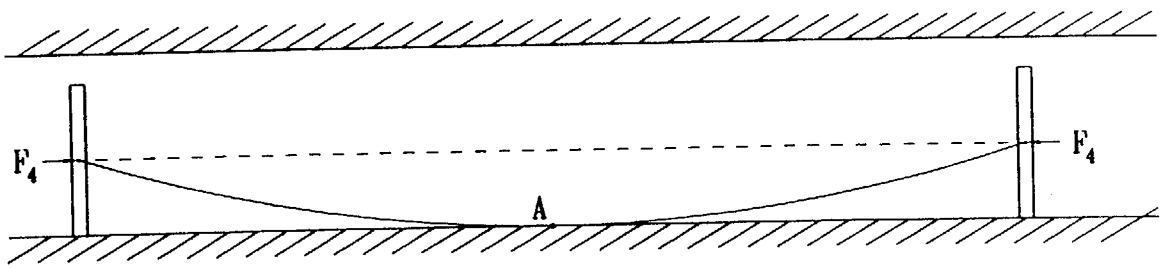


FIGURE 3.5.5; POINT CONTACT AT LOAD F_4
MAXIMUM BENDING MOMENT AT A

FUEL ROD DEFORMATION PHASES

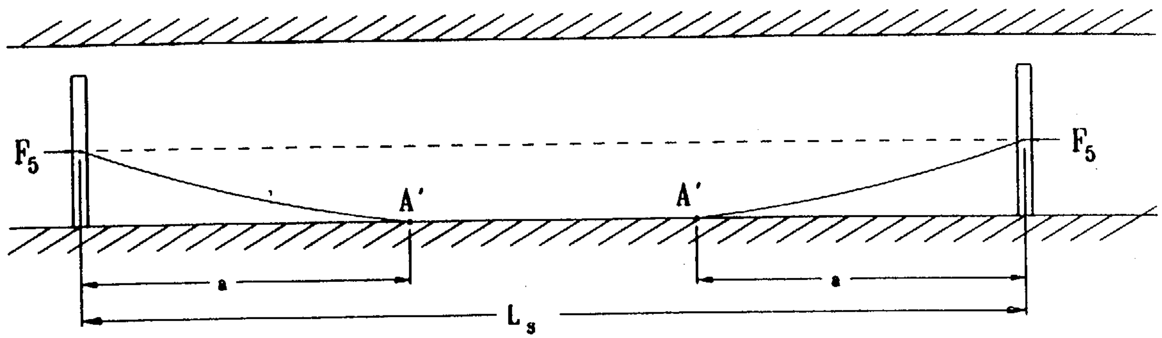


FIGURE 3.5.6; EXTENDED REGION OF CONTACT
$F_5 > F_4$, ZERO BENDING MOMENT AT A'

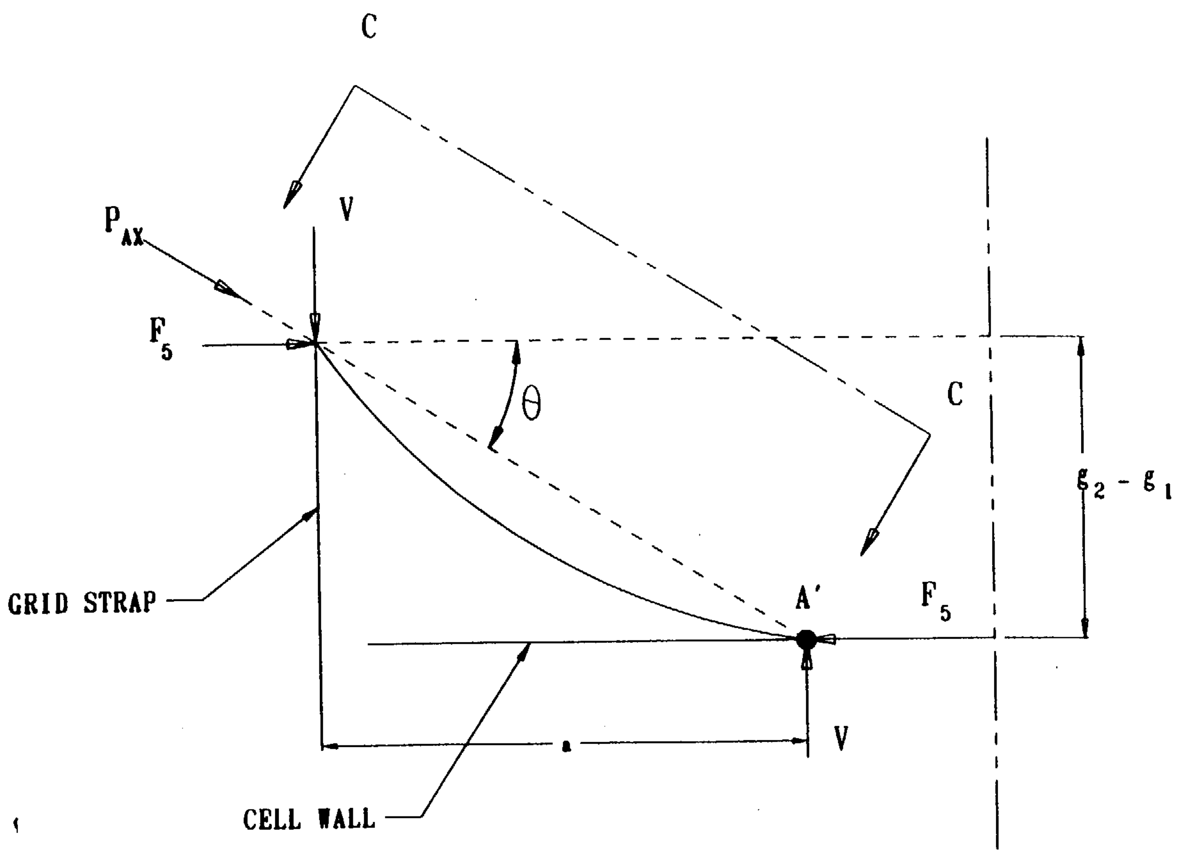
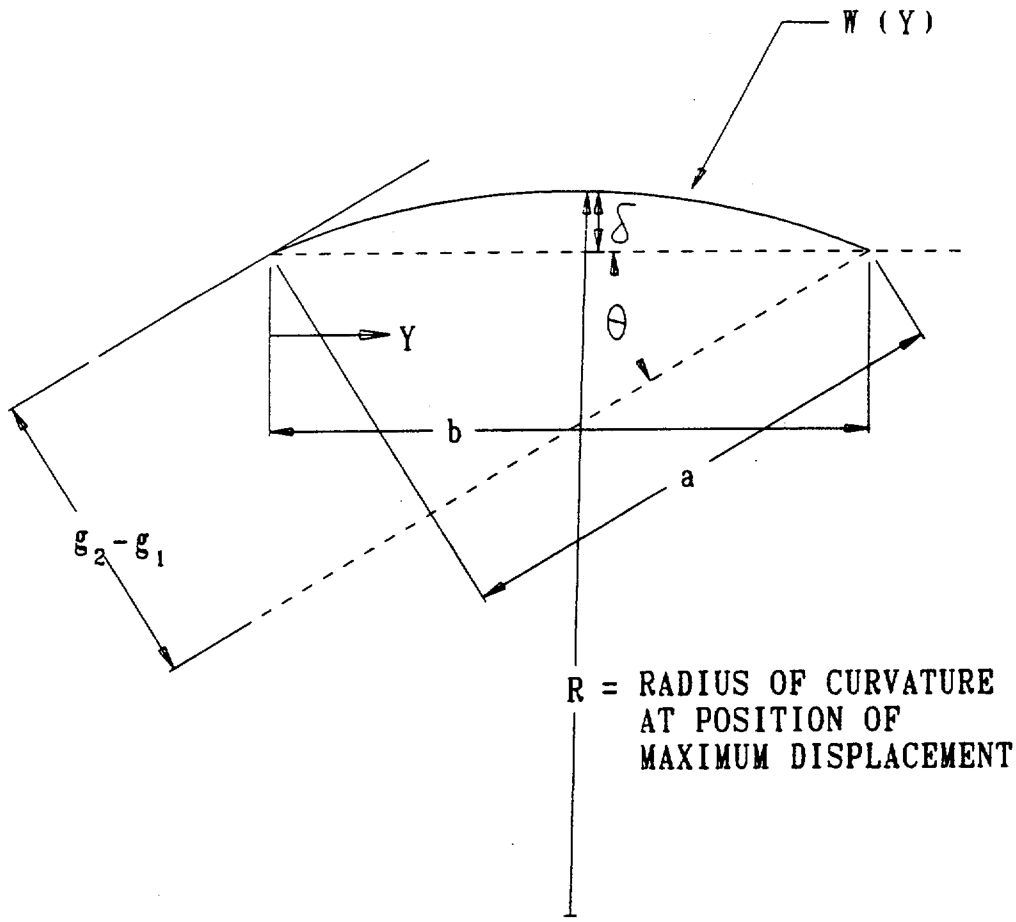


FIGURE 3.5.7; FREE BODY DIAGRAM WHEN MOMENT AT A' = 0
$P_{AX} = F_5 / \cos(\theta)$. RESISTING MOMENT M_R
AT GRID STRAP NOT SHOWN



$$Z = R - \delta$$

$$W(Y) = \delta \, SIN(\pi Y/b)$$

FIGURE 3.5.8; VIEW C - C

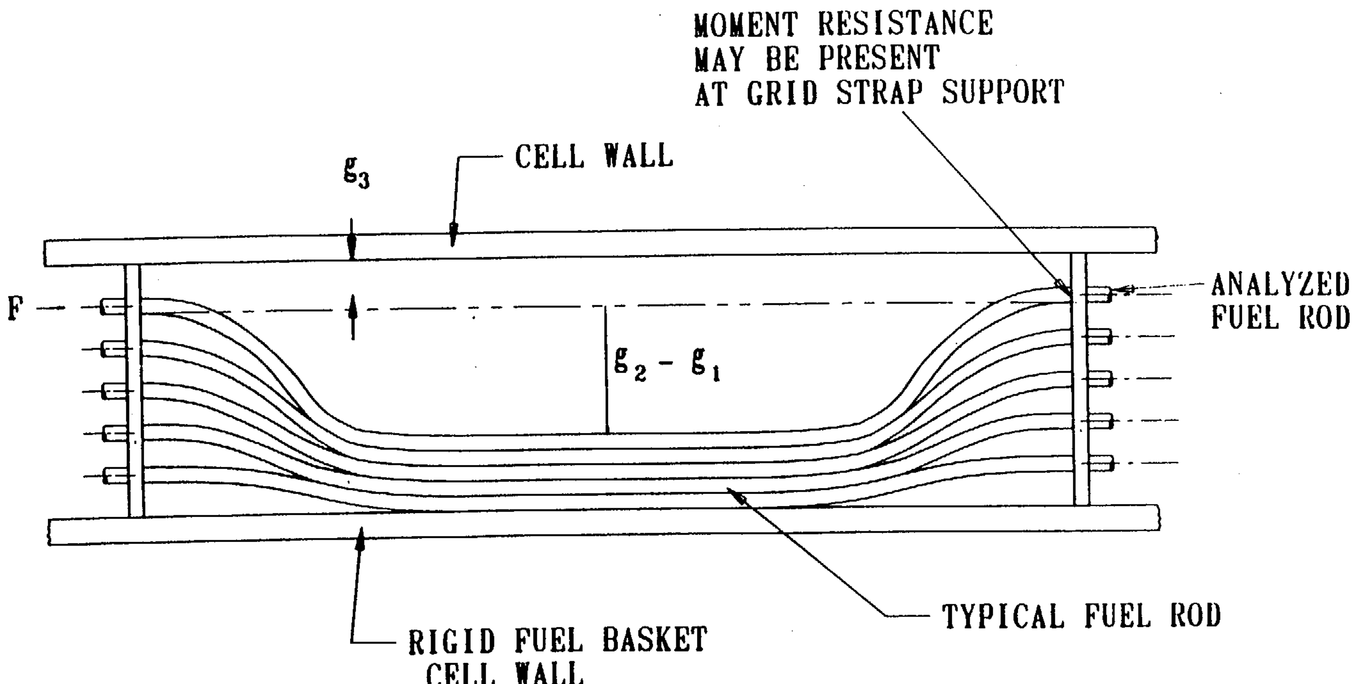


FIGURE 3.5.9; EXAGGERATED DETAIL SHOWING MULTIPLE FUEL RODS SUBJECT TO LATERAL DEFLECTION WITH FINAL STACKING OF ROD COLUMN

NUCLEAR FUEL MANAGEMENT

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APPENDIX C

TYPICAL NUCLEAR POWER REACTOR DATA

General data	PWR (W)	PWR (B & W)	PWR (CE)	BWR/6	HTGR	LMFBR	GCFR	CANDU PHW
Fuel assemblies								
Type	Square bundles	Square bundles	Square bundles	Canned-square bundles	Hexagonal graphite prisms	Hexagonal canned bundles	Hexagonal canned bundles	Pressure tube bundles
Number of assemblies	193	205	241	732	3944	394	347	473
Fuel element array	17 × 17	17 × 17	16 × 16	8 × 8	132 pins	hex	hex	pressure tubes
Assembly dimension (cm)	21.4 × 21.4	21.7 × 21.7	20.3 × 20.3	14 × 14	35 × 79	12 × 12	17 × 17	8 × 50
Assembly pitch (cm)	21.5	21.8	20.7	30.5	36.1	12.4	17.5	27.9
Number of fuel elements/assembly	264	264	236	63	132	217	225	28
Total number of fuel locations	50,952	54,120	56,876	46,116	35,496	85,464	77,031	13,244
Fuel element data								
Type	Clad rod	Clad rod	Clad rod	Clad rod	Graphite UC, ThO_2 rod	Wire-wrap clad rod	Vented clad rod	Clad rod
Fuel element pitch (cm)	1.25	1.27	1.28	1.62		0.725	1.14	1.65
Fuel element O. D. (cm)	0.94	0.96	0.97	1.25	1.56	0.579	0.805	1.52
Pitch/diameter	1.32	1.32	1.33	1.30		1.25	1.41	1.08
Clad thickness (cm)	0.0572	0.0597	0.0635	0.0864		0.038	0.0295	0.038
Fuel pellet diameter (cm)	0.819	0.823	0.825	1.056	1.56	0.66	0.739	1.44
Pellet–clad gap (cm)	0.0082	0.010	0.0089	0.008		0.012	0.012	
Fuel enrichment	2.1/2.6/3.1	2.91	1.9/2.4/2.9	2.2–2.7	93.5	10–15	10–15	nat U