ENCLOSURE 1

EVALUATION OF THE ERROR AND PROPOSED PATH FORWARD

1.0 Description of Error

The analysis in SARP Vol. I, Chapter 3, Appendix B, is a one-dimensional thermal analysis which predicts the percentage of the fuel-bearing material that would be melted if a fuel element were directly exposed to the hypothetical accident condition (HAC) fire without benefit of the container. Chapter 4, *Containment*, uses the results from this analysis as a partial basis for concluding that the regulatory release limits (per 10 CFR 71.51 and Appendix A) would not be exceeded. The analysis predicts that approximately 13% of the inner or outer fuel element fuel-bearing material would be melted under the assumed conditions.

The analysis in Chapter 3, Appendix B, incorrectly assumed the pre-machined outer side plate thicknesses rather than the final thicknesses shown on the fuel element assembly drawings in Chapter 1 of the SARP (Drawings E-42118 and E-42126, included as Figures 1.7 and 1.8, respectively). The pre-machined outer side plate thickness of the outer fuel element is 0.408 in., whereas the final thickness is actually 0.128 in. (minimum). Similarly, the pre-machined outer side plate thickness of the inner fuel element was assumed to be 0.377 in. in the analysis, whereas the final thickness is 0.105 in. (minimum). Substituting the correct thicknesses in the existing analysis, and revising the initial starting temperature of the analysis to correspond to the maximum normal conditions of transport (NCT) package temperature (consistent with the requirements of 10 CFR 71), results in predicted melting of approximately 22% and 24% for the inner and outer elements, respectively.

2.0 Identification of Necessary Operational Restrictions

Following the determination that the identified error resulted in non-conservative changes to the Chapter 3, Appendix B, analysis notification of the error was issued to registered users of the package and use of the package was suspended pending further evaluation. No 5797 packages were in transit when the error was identified. In accordance with 10 CFR 71.7(b), the Department of Energy notified the NRC by letter dated March 5, 2020 (Reference 1). The NRC has since provided ORNL with a Letter of Authorization, dated March 18, 2020 (Reference 2), which authorizes the use of the 5797 package until July 31, 2020.

3.0 Path Forward

ORNL has revised the SARP analyses to correct the identified error by determining (1) the amount of melting that would be expected given the corrected fuel side plate thicknesses; (2) the currently applicable A_2 value, per 10 CFR 71, Appendix A; (3) the activities of the inner and outer fuel elements; and (4) the minimum percentage of fuel-bearing material that must melt in order to exceed the regulatory limit of an A_2 value in one week. The results are summarized below:

(1) Using the correct outer side plate thicknesses in the calculations in the appendix, as well as revising the initial starting temperature of the analysis to correspond to the maximum NCT package temperature, the percentage of fuel-bearing material expected to be melted/exposed would be approximately 24.4% and 21.8% for the outer and inner fuel elements, respectively.

(2) Unirradiated HFIR fuel contains U-234, U-235, U-236, and U-238. The A_2 values for U-235 and U-238, per 10 CFR 71, Appendix A, are unlimited. The sum of fractions method for mixed normal form material specified in 10 CFR 71, Appendix A, is used to calculate the appropriate A_2 value. For HFIR fuel, the A_2 value is 1.65E-01 Ci (6.105E-03 Tbq).

(3) The activities of the inner and outer fuel elements are derived using the nominal fuel contents presented in Chapter 1, along with the given limits for U-235 in the inner and outer elements of 2630g and 6880g, respectively.

(4) Using the currently applicable A_2 value, it was found that greater than 90% of the fuel-bearing material in the inner fuel element, or 34% of the fuel-bearing material in the outer fuel element, must melt before the regulatory limit would be exceeded. Therefore, the postulated thermal hypothetical accident conditions would not result in a release of an A_2 value in one week.

An assessment of the implications of the identified error has been completed which identifies the sections of the SARP that need to be revised. The affected chapters include Chapters 2, *Structural Evaluation*; Chapter 3, *Thermal Evaluation*; and Chapter 4, *Containment*. The description of the package in Chapter 1, *General Information*, does not utilize the incorrect final side plate thicknesses or the results of the HAC thermal test. The structural analysis in Chapter 2, and the analysis in Chapter 6, *Criticality Evaluation*, reference the fuel assembly drawings with the correct outer side plate thicknesses. Additionally, the correct outer side plate thicknesses for both inner and outer fuel elements are explicitly listed as assumed dimensions in Chapter 2, Appendix C. Minor changes to Sections 2.1.2 and 2.7.3 were made to be consistent with the updated analysis and current regulations.

The base analysis in Chapter 3 is an antecedent to the analysis in Appendix B of that chapter and does not rely upon its results, except for reporting those results in the summary of the HAC thermal evaluation in Section 3.5. Similar summaries of the results from Chapter 3, Appendix B, appear in Chapter 4. The affected sections have been updated and the page changes are provided in an enclosure to this letter. Chapters 7 through 9 are unaffected by the error or the subsequent changes to the SARP.

4.0 References

1. Letter, J.M. Shuler to J.B. McKirgan, *71-5797 SAR Error Notice and Request for Continued Use of HFIR Outer Container*, U.S. Department of Energy, March 5, 2020.
2. Letter, J.B. McKirgan to J.M. Shuler, *Authorization for Shipments Using the Model No. Inner HFIR Unirradiated Fuel Element Shipping Container, and Outer HFIR Unirradiated Fuel Element Shipping Container*, U.S. Nuclear Regulatory Commission, March 18, 2020.

ENCLOSURE 2

SUMMARY OF CHANGES TO SAFETY ANALYSIS REPORT FOR PACKAGING: THE ORNL HFIR UNIRRADIATED FUEL ELEMENT SHIPPING CONTAINER

Summary of Volume I Changes*		
SARP Location	Change	Reason for Change
Cover page and List of Effective Pages (LOEP)	Updated for Rev. 12.	Editorial change reflecting latest revision.
Revision Log, pg vii-a thru vii-g	Updated for Rev. 12.	Editorial change explaining scope of Rev. 12.
Section 2.1.2, pg. 2-2	Changed "0.1 Ci (A_2 quantity)" in first full paragraph to "an A_2 quantity".	0.1 Ci corresponds to the previous requirements in 10 CFR 71 for calculating A_2 for uranium enriched to > 20%. The current requirements in 10 CFR 71, Appendix A, for mixed normal form material, applied to HFIR fresh fuel, gives an A_2 value of 0.165 Ci, when calculated as described below (see reason for change to Section 4.1, pg. 4-3).
Section 2.7.3, pg. 2-24	Revised the second sentence of the section from "…it was shown that only 13% of the fuel-bearing portion of a HFIR fuel assembly will melt if the assembly is exposed to the regulatory fire for 30 minutes" to "…it was shown that less than an A_2 amount of radioactive material will be released if the assembly is exposed to the regulatory fire for 30 minutes."	This section has been updated for a new analysis which shows that the amount of fuel which melts in the regulatory fire is more than previously assumed, while the amount which can be melted without potentially releasing an A_2 amount of material is also more than previously assumed.
Section 3.5, pg. 3-7	Revised the third sentence of the fourth paragraph in Section 3.5 from "…show that less than 15% of fuel bearing portion of a HFIR fuel assembly will melt if the assembly is exposed to a 1475°F fire for 30 minutes" to "…show that less than an A_2 amount of radioactive material will be released if the assembly is exposed to a 1475°F fire for 30 minutes."	This section has been updated for a new analysis which shows that the amount of fuel which melts in the regulatory fire is more than previously assumed, while the amount which can be melted without releasing an A_2 amount of material is also more than previously assumed.

Chapter 3, Appendix B, List of Effective Pages, pg. 3-B-1	Updated revision level for pages 3-B-1, B-1 through B-4, B-8, and B-9.	These pages in Chapter 3, Appendix B, were revised in Rev. 12.
Chapter 3, Appendix B, Summary, pg. B-1	Changed the predicted amount of melted fuel in fourth sentence.	This section has been updated for a new analysis which shows that the amount of fuel which melts in the regulatory fire is more than previously assumed.
Chapter 3, Appendix B, pgs. B-2 through B-4 and B-8	Revised the applicable portions of the calculation section of Chapter 3, Appendix B, and provided references to the correct drawings for side plate dimensions for the inner and outer fuel elements. Updated the conclusion of calculation analysis to reflect that the Appendix B test now begins at the NCT maximum package temperature of 169ºF.	The incorrect side plate thicknesses were updated with the correct values, and the subsequent calculations in this section were updated to reflect the changes. Using the NCT (169ºF) conditions for the HAC thermal test is consistent with current regulation specified in 10 CFR 71.73 (b).
Chapter 3, Appendix B, References, pg. B-9	References to the manufacturing side plate drawings were deleted. References for the side plate dimensions now call out the fuel drawings in Chapter 1.	The assembly drawings found in SARP Chapter 1 provide the correct final side plate dimensions.
Section 4.1, pg. 4-3 and (new) pg. 4-3a	Discussions of the A_2 values associated with the fuel material for the inner and outer fuel elements were updated to reflect the current regulations.	There is no longer an A_2 value for uranium enriched to > 20%. The appropriate A_2 value was derived using the sum of fractions method for mixtures of normal form material, per 10 CFR 71, Appendix A.

Section 4.3.3, pgs. 4-8 and 4-9	Section 4.3.3 was updated to reflect that, at a minimum, releases of greater than 90% of the fuel-bearing material from an inner element and greater than 34% of the fuel-bearing material from an outer element are necessary to release an A_2 amount of material. Additionally, the second paragraph was updated to reflect that the analysis in Chapter 3, Appendix B, concludes that less than 25% of the fuel-bearing material from a fuel element will melt in the HAC thermal test when the container is not present.	This section has been updated for a new analysis which shows that the amount of fuel which melts in the regulatory fire is greater than previously assumed, while the amount which can be melted without releasing an A_2 amount of material is also greater than previously assumed.

*No changes were required to Vol. 2 of the SARP. Its revision level was increased to remain consistent with Vol. 1.

ENCLOSURE 3

SAFETY ANALYSIS REPORT FOR PACKAGING THE ORNL HFIR UNIRRADIATED FUEL ELEMENT SHIPPING CONTAINER

VOLUME 1

ORNL/TM-V11656/V1
Rev. 12

SAFETY ANALYSIS REPORT FOR PACKAGING THE ORNL HFIR UNIRRADIATED FUEL ELEMENT SHIPPING CONTAINER VOLUME I

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LIST OF EFFECTIVE PAGES

Volume I

Page	Revision
Cover Page	12
ii thru iv (Table of Contents)	8/7/98
v (List of Figures)	8/7/98
vi (List of Tables)	8/7/98
vii-a thru vii-g (Summary of Revision 12 changes)	12
viii (Abstract)	8
1-1	11
1-2	11
1-3	5
1-4	6
1-5	11
1-6 thru 1-7	5
1-8 thru 1-9	10
1-10	9
1-11	5
1-12 thru 1-13	10
1-14 thru 1-17	4
1-18	5
1-19	10
1-20	4
1-21	5
1-22	10
1-23	4
1-24	10
2-1	4
2-2	12
2-3	5
2-4	4
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2-8	5
2-9	4
2-10	2
2-11	4
2-12	2
2-13 thru 2-14	5
2-15	4
2-16 thru 2-21	5
2-22	7

LIST OF EFFECTIVE PAGES

Volume I

2-23	5
2-24	12
2-25 thru 2-26	4
2-27	7
2-28	4
2-29 thru 2-29a	7
2-30 thru 2-31	4
2-A-1	4
2-B-1	4
2-C-1	11
2-D-1	4
2-E-1	5
2-F-1	5
3-1	4
3-2 thru 3-3	1
3-4 thru 3-6, and 3-7a	4
3-7	12
3-8 thru 3-12	1
3-13	4
3-14	1
3-15 thru 3-20	4
3-21	11
3-22	4
3-23	1
3-A-1	11
3-B-1	12
4-1 thru 4-2	4
4-3 thru 4-3a	12
4-4 thru 4-7, and 4-10 thru 4-11	4
4-8 thru 4-9	12
4-12	10
4-13	4
4-A-1	4
4-B-1	4
4-C-1	10
4-D-1	4

Note: All appendices in Chapter 4 contain reference documents. These reference documents do not reflect the revision number control system of the SAR since they were issued as separate documents.

5-1	6
5-2 thru 5-3	4

Revision Log

Revision	Description of Changes	Approved by NRC Certificate of Compliance (CoC) Revision Number:
12	• Rev. 12 of the SARP addresses a latent error found in Appendix B of the Chapter 3 thermal analysis. The amount of melting predicted to occur in this postulated scenario was skewed by the use of pre-machined HFIR fuel element side plate thicknesses in the calculations rather than the final assembly dimensions. Additionally, the A_2 value previously calculated has been updated to the latest regulations. Vol. II of the SARP is not affected by these changes; thus, the only change in Vol. II is to the revision level. • The Cover Page and List of Effective Pages (LOEPs) for both Volume 1 and Volume 2 were updated to Revision 12. The LOEPs were updated to identify pages changed in this revision as Revision 12. • Updated the Revision Log for Revision 12 changes. • Section 2.1.2: Revised language in first full paragraph of pg. 2-2 from “0.1 Ci (A_2 quantity)” to “an A_2 quantity” since the previous A_2 quantity is no longer applicable under current regulations. • Section 2.7.3: Revised the second sentence of the section to state “…less than an A_2 amount of radioactive material will be released.” This section has been updated for the revised analysis which corrects a latent error in Chapter 3, Appendix B, and updates the A_2 value to be consistent with current regulations. • Section 3.5: Revised the third sentence of the fourth paragraph in Section 3.5 (pg. 3-7) to state “…an A_2 amount of radioactive material will be released...”. • Ch. 3, Appendix B, LOEP, pg. 3-B-1: Editorial change to update revision level for pages 3-B-1, B-1 through B-4, B-8, and B-9. • Ch. 3, Appendix B, Summary, pg. B-1: Changed the predicted amount of melted fuel in the fourth sentence to reflect the revised analysis in Chapter 3, Appendix B. • Ch. 3, Appendix B, pgs. B-2 through B-4 and B-8: Revised the applicable portions of the calculation section of Ch. 3, Appendix B, and provided references to the correct drawings for side plate dimensions for the inner and outer	To Be Determined* * CoC revision number will be included upon approval by the NRC.

Revision	Description of Changes	Approved by NRC Certificate of Compliance (CoC) Revision Number:
	fuel elements. Updated the conclusion of calculation analysis to reference to reflect that the Appendix B test now begins at the NCT maximum temperature of 169ºF. • Ch. 3, Appendix B, References, pg. B-9: The references to the manufacturing side plate drawings were deleted. References for the side plate dimensions now call out the fuel assembly drawings in Chapter 1. • Section 4.1, pg. 4-3 and (new) pg. 4-3a: Discussions of the fuel material and associated A_2 values for the inner and outer fuel elements were updated to reflect the current regulations. • Section 4.3.3, pgs. 4-8 and 4-9: This section was updated to reflect that, at a minimum, releases of greater than 90 % of the fuel from an inner element and greater than 34% of the fuel from an outer element are necessary to release an A_2 amount of material. Additionally, the second paragraph was updated to reflect that the analysis in Ch. 3, Appendix. B, concludes that less than 25% of the fuel-bearing material from a fuel element will melt in the HAC thermal test when the container is not present.	
11	• The Cover Page and LOEPs for both Volume 1 and Volume 2 were updated to Revision 11. The LOEPs were updated to identify pages changed in this revision as Revision 11. • The Summary of Revision page in Volume 1 was replaced with a detailed revision log in Volume 1. The Summary of Revision page in Volume 2 was revised to refer the reader to the detailed revision log in Volume 1. • Section 1.1 and the introductory paragraph in Chapter 6 were revised to clarify that since current regulations no longer use Fissile Classes. A Criticality Safety Index of 0.4 is specified in the CoC. • An editorial change was made to Section 1.1 to clarify that the containers comply with regulations in effect as of the date of the original application. • Section 1.2.1, a typographical error was corrected related to the wood post glued to the inside wood base 4 ¼ inches from the center of the cask, as this dimension is shown on drawing M-20978-EL-002, Rev. E (included as Figure 1.3)	20

<table>
<tr><th>Revision</th><th>Description of Changes</th><th>Approved by NRC Certificate of Compliance (CoC) Revision Number:</th></tr>
<tr><td></td><td>
• Table 1.1 was revised to correct the drawing number listed for the HFIR Outer Fuel Element – Outer Side Plate.
• Figure E-20 in Appendix E of Chapter 2 was replaced with a copy that was redrawn to improve legibility. Revision level indicated for this figure was not changed because no technical changes were made in the figure.
• The LOEPs for Appendix C of Chapter 2 (page 2-C-1) was revised to change the indicated revision level of page 42 from Rev 4 to Rev 7. Page 42 was previously revised as part of the revision 7 update, but the indicated revision level was inadvertently left at Rev 4 on the LOEPs.
• The microfiche previously included as Appendix 3.6.A was replaced with a printout of the microfiche contents to facilitate conversion of the document to an electronic format. The associated title page for the appendix was revised to delete reference to the microfiche.
• Photos 1-6 in Chapter 4, Appendix A were replaced with more legible copies. Revision level indicated for this appendix was not changed because no content changes were made in the appendix.
• Appendices B and D in Chapter 4 were replaced with more legible copies. Revision level indicated for these appendices was not changed because no content changes were made in the appendices.
• Chapter 9, Section 9.1, Introduction, first paragraph, added wording to clarify that users of the containers must be registered Users and must maintain an acceptable Quality program.
• Editorial changes were incorporated in Chapter 9 to reflect current organizational structure, current document identifiers, and make minor clarifications.
</td><td></td></tr>
<tr><td colspan="2">CoC Rev 19 made editorial changes in the CoC, and no changes to the SARP.</td><td>19</td></tr>
<tr><td>10</td><td>
• Volume 1 and Volume 2 cover pages were revised to update the revision level and date.
• The lists of effective pages for Volumes 1 and 2 were revised to reflect the changes in Revision 10.
• The “summary of changes” page for each volume was revised to list changes in revision 10 of the document.
</td><td>18</td></tr>
</table>

Revision	Description of Changes	Approved by NRC Certificate of Compliance (CoC) Revision Number:
	• Sections 1.2.1.1, 1.4, 4.5, 9.3.3, 9.3.5, 9.3.6, and 9.3.9 were revised to cite the most recent revision of the fuel element specification, RRD-FE-3. • Table 1.1 was revised to cite the most recent revision of the drawings listed in the table. • Drawings M-20978-EL-002E and M-20978-EL-003E, included as Figures 1.3 and 1.4 were revised to include the trefoil label required by 49 CFR 172 Appendix B. • Drawings E-42118, E-42126, D-42114, and D-42122, included as Figures 1.7, 1.8, 1.14, and 1.17 respectively, were updated to incorporate editorial changes and to reflect the results of improved inspection techniques implemented at the fuel fabricator. • Chapter 4, Appendix C was revised to include the most recent revision of RRD-FE-3, the fuel element specification.	
9	• Volume 1 and Volume 2 cover pages were revised to update the revision level and date. • The lists of effective pages for Volumes 1 and 2 were revised to reflect the changes in Revision 9. • The “summary of changes” page for each volume was revised to list changes in revision 9 of the document. • Drawings M-20978-EL-002E and M-20978-EL-003E, included as Figures 1.3 and 1.4 respectively, were revised to include new plywood specifications that facilitate maintenance on the containers. *(CoC Rev 14 included approval of changes in the plywood specification shown on drawings M-20978-EL-002E and -003E.)* • Drawing M-20978-EL-008E, included as Figure 1.5, was revised to update nameplate information. (*CoC Rev. 13 included approval of nameplate changes on drawing M-20978-EL-008E. CoC Rev. 12 included reference to the September 1996 submittal, however it did not include approval of the change.*) • Drawings D-42114 and D-42122, included as Figures 1.14 and 1.17 respectively, were revised to change the maximum allowable distance between the fuel area and the edge of the fuel element plate. These changes were based on use of more sensitive radiographic inspection techniques by the fuel fabricator. (*CoC Rev 17 included approval of changes*	17, 16, 14, 13, 12

Revision	Description of Changes	Approved by NRC Certificate of Compliance (CoC) Revision Number:
	in the allowable distance between the fuel area and edge of the fuel element plate as shown on drawings D-42114 and D-42122. These drawing changes were originally submitted in August 2007. Revision 16 of the CoC cites the August 2007 submittal but the revised drawings were not listed until Revision 17 of the CoC.)	
	CoC Rev 15 made no changes to the SARP.	15
8	• Volume 1 cover page was revised to update the revision level and date. • The list of effective pages for Volume 1 was revised to reflect the changes in Revision 8. • The "summary of changes" page for Volume 1 was revised to list changes in revision 8 of the document. (No changes were made to Volume 2.) • Minor editorial changes were made in the Table of Contents, List of Figures, List of Tables, and Abstract to address discrepancies identified by an internal self-assessment. • Drawings E-42118 and E-42126, included as Figures 1.7 and 1.8 respectively, were revised to change reference to drawings: M-11524-04-101-D and M-11524-04-102-D to M-11524-OH-101-D and M-11524-OH-101-D. (*CoC Rev 13 included approval of these drawing changes – CoC Rev 12 included reference to the request for this change but did not include approval of the change)*	13, 12
	CoC Rev. 11, no changes were made to the SARP	11

<table>
<tr><th>Revision</th><th>Description of Changes</th><th>Approved by NRC Certificate of Compliance (CoC) Revision Number:</th></tr>
<tr><td>7</td><td>Volume 1 and Volume 2 cover pages were revised to update the revision level and date.
The lists of effective pages for Volumes 1 and 2 were revised to reflect the changes in Revision 7.
The following changes were made to incorporate clarifications included in the DOE letter from Michael Wangler to Charles MacDonald of NRC, dated February 26, 1992:
Table 2.1 was revised to include bolting material properties
Section 2.7.1.2 and Section 6.4 of Appendix C were revised to clarify acceptable strength of the closure bolts.
Section 2.7.6 was revised to include a summary of cumulative damage.
Table 6.2 was revised to add a value for density for the Al cermet.</td><td>9</td></tr>
<tr><td>6</td><td>Sections 1.2.3, 4.1, 5.2, and Table 6.2 were revised to change the maximum U^{235} content from 2.6 kg to 2.63 kg and from 6.8 kg to 6.88 kg for the inner and outer fuel element containers, respectively.
An existing supplement was added to the fuel element specification, included as Chapter 4 Appendix C, to ensure that all fabrication documentation is included in the SARP. The supplement provided specifications for U_3O_8 provided by ORNL to the fuel fabricator.</td><td>10</td></tr>
<tr><td>5</td><td>Revision 5 was the first revision of the SARP submitted for NRC review. A detailed description of the changes incorporated in this revision is not available. The changes included in revision 5 addressed the final set of comments from the DOE review in preparation for submitting the document to the NRC for upgrading the CoC from B( )F to B(U)F. The following pages were changed in revision 5:
Chapter 1
1-1 through 1-4; 1-6; 1-7; 1-11; 1-18; 1-19; 1-21; & 1-22.
Chapter 2
2-2; 2-3; 2-5 through 2-8; 2-13; 2-14; 2-16 through 2-24; 2-27; selected pages in Appendix C, selected pages in Appendix E; and selected pages in Appendix F.</td><td>9, 8</td></tr>
</table>

Revision	Description of Changes	Approved by NRC Certificate of Compliance (CoC) Revision Number:
	(CoC Rev 8 invoked the maintenance program and operating procedures from ORNL/TM-11656 (submitted with May 30, 1991 DOE letter), but did not invoke new bolt specification, etc. for upgrade to meet B(U)F. CoC Rev 8, and prior revisions cited ORNL/ENG/TM-9 as the applicable safety analysis. *CoC Rev 9 upgraded the designation from B( )F to B(U)F. Drawings included in SARP, Rev. 5, Chapter 1 cited correct bolt specifications and blocking device. Procedures described in SARP, Rev. 5, Chapter 7 included blocking device. With the upgrade to B(U)F designation, references to ORNL/ENG/TM-9 were removed from the CoC and replaced with ORNL/TM-11656.)*	
4 – 0	This document was prepared to demonstrate compliance with then-current regulations and allow upgrading the CoC from B( ) to B(U)F. Revisions 0 through 4 were reviewed by DOE, but not by the NRC. Revisions 1 through 4 addressed comments that resulted from DOE review of the SARP in preparation for submittal to NRC. A detailed description of the changes incorporated in revisions 1 through 4 is not available. See the ORNL letter from H. A. Glovier to H. Randall Fair of DOE, dated April 26, 1991, Subject: “Responses to DOE-HQ Evaluation of Q0/Q1/Q2 Questions on the DOE Certificate of Compliance (No. 5797) License Renewal Request for the High Flux Isotope Reactor Unirradiated Fuel Shipping Containers” for submittal of SARP revision 4 changes.	Not applicable

specified in 10 CFR 71.71 there should be no loss or dispersal of radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging.

For hypothetical accident conditions, the applicable criteria are given in 10 CFR 71.51(a) (2); namely, that under the loading conditions specified in 10 CFR 71.73 there should be no escape of radioactive material exceeding an A2 quantity in one week and no external radiation dose rate exceeding 1 rem per hour at 1 m from the external surface of the package. These criteria are satisfied if the fuel element does not melt or fracture into numerous small pieces.

The design criteria are based on the importance of a particular structural component to the performance of the total packaging under both normal and hypothetical conditions of transport. As stated previously, the fuel element is the critical component. The fuel element is composed of fuel plates (which contain the fuel) and side plates. (Refer to Chapter 1, Fig. 1.11 for a photograph of the element. Additional detailed drawings are also provided in Figs. 1.12 through 1.17.) The side plates provide a cylindrical shell for the fuel plate assembly and extend a minimum of approximately 2.5 in. past the fuel plates on either end of the fuel element assembly. The composition of the fuel plates is such that 2 in. of solid aluminum is provided on either end of the fuel plate.

The fuel element is designed to meet the containment requirements for both the normal and hypothetical accident conditions as stated in 10 CFR 71.51 (a) (1 and 2). The analysis in Chapter 3 shows that the fuel element can be subjected to the regulatory fire, without the container, and still maintain the acceptable A2 release values. Chapter 4 discusses containment boundaries for the fuel element. The primary containment boundary is defined as the cladding bonded to the fuel matrix of the HFIR fuel plates.

Each end of the fuel element has a minimum of approximately 4.5 in. of solid aluminum that could be deformed without affecting the containment of the fuel. Because small deformation on the ends of the side plates would not adversely affect the integrity of the fuel plate, the stress on the side plates can exceed the yield stress of the aluminum. However, the stress in the fuel plate should be below or approximately at the yield stress. Local yielding of the fuel plates is acceptable provided the overall geometry changes in the fuel element do not affect the criticality calculations. Local yielding is allowable because testing has been performed on irradiated fuel plates that shows the plates capable of bending in excess of 80 degrees without cracks being formed in the plate. However, local bending of fuel plates shall be limited to less than 20 degrees, which provides a factor of safety greater than 4.

Additional criteria for other components of the fuel element container are as follows:

1. The lid should remain secured to the body of the container during normal transport conditions.

This section makes reference to the results of tests performed on plutonium nitrate shipping containers that were reported[14] in the June 1965 *Proceedings of the International Symposium for the Packaging and Transportation of Radioactive Materials*. Puncture tests were made on these plutonium containers during which the containers were dropped 40 in. onto a 6-in.-diameter steel bar. The resulting damage was minimal; no ruptures occurred.

A comparison of the HFIR inner and outer packages with the plutonium nitrate packages is given in Table 2.3. The packages are similar in that both the HFIR and plutonium nitrate packages make use of low-carbon steel drums as the outer container and use wood as the thermal insulator. The payload weights of the packages differ significantly. The 3-1iter plutonium nitrate payload weighs approximately 10% of the HFIR inner fuel assembly; the 10-1iter plutonium nitrate payload weighs approximately 15% of the HFIR outer fuel assembly. The container is shown in Fig. 2.6.

The lid closures on the two types of packages are also significantly different. The bolt-type locking ring used to secure the covers of the plutonium nitrate packages is less secure than the bolted closure design of the HFIR packages. Failure of the single locking ring bolt under direct impact could allow the cover to slip off under load.

With regard to the potential puncture of the packaging as a result of a 40-in. drop onto a 6-in.-diameter steel bar, local stiffness of the packaging is most important. The thickness and stiffness (modulus of elasticity and crush strength) of the wood are approximately the same for both types of packages. However, the thickness of the carbon steel shell and top and bottom plates of the plutonium nitrate packages is one-half the thickness of that of the HFIR packages. Hence, the plutonium nitrate packages are more likely to be punctured during such a test. Because no ruptures occurred and damage resulting from the test was minimal, it was concluded that the puncture test would produce minimal damage of the HFIR packages.

2.7.3 Thermal

Calculations were prepared for the melting of inner and outer HFIR fuel elements in a 1475° F (800° C) fire for the case in which the element is completely removed from the shipping cask and directly exposed to the fire. Based on the conservative evaluation presented in Appendix B to Chapter 3, it was shown that less than an A_2 amount of radioactive material will be released if the assembly is exposed to the regulatory fire for 30 minutes. Based on these results, which are in compliance with the regulations, the structural integrity of the fuel container is not essential to meeting the thermal requirements of the regulation.

3.4.5 Maximum Thermal Stress

No thermal gradients are sufficient to cause significant thermal stresses in metallic members.

3.4.6 Evaluation of Package Performance for Normal Conditions of Transport

The previous assessments and over 20 years of in-service operation demonstrates that the package is adequate for normal conditions of transport.

3.5 Hypothetical Accident Thermal Evaluation

The damage from the free-drop and puncture portions of the hypothetical accident should not adversely affect the performance of the container in the hypothetical thermal accident. This statement is based on a comparison to container USA/9853/B(U)F.[15] Damage from the free-drop and puncture portions of the hypothetical accident gives lower temperatures than the undamaged case.

Also, if the assumption is made that the foam crushes totally and the plywood deforms as calculated in Chapter 2, the temperature of the inner wood would be comparable to a case evaluated in the thermal analyses. This case assumed no foam and resulted in the inner wood temperature being less than 290° F. Hence, the undamaged configuration was assumed.

The HEATING computer program, version 6.1[6] was used to determine the temperature distribution in the container that would result from the prescribed 30-min thermal exposure. The material constants given in Table 3.1 were used in the input to HEATING.

A bounding "worst case" thermal analysis was performed and is documented in the Appendix B to this chapter. In this worst case, the inner and outer elements were totally exposed to the regulatory fire without the benefit of any fuel container around the elements. The results of this conservative analysis show that less than an A_2 amount of radioactive material will be released if the assembly is exposed to a 1475°F fire for 30 minutes. It is emphasized that the analysis located in Appendix B is a very conservative bounding calculation which demonstrates that the fuel is within the acceptable limits of melting, even if the container is nonexistent during the regulatory fire.

APPENDIX B

(Chapter 3)

HFIR UNIRRADIATED FUEL SHIPPING CASK FIRE – CALCULATIONS OF WORST-CASE MELTING OF FUEL ELEMENTS

List of Effective Pages

Page	**Revision**
B-1 through B-4	12
B-5 through B-7	08/90
B-8 & B-9	12

SUMMARY

These calculations were prepared for the melting of inner and outer HFIR fuel elements in a 1475° F (800° C) fire for the case in which the element is completely removed from the shipping cask and directly exposed to the fire. It should be emphasized that this is not considered a very likely accident because all indications are that the elements will not be ejected from the casks. The calculations are approximations with a large degree of conservatism. The estimate is that less than 25% of the uranium-bearing fuel would be melted for this scenario. U_3O_8 decomposes to UO_2 at 1300° C (2372° F), which in turn melts at 2500° C (4532° F). Thus, the hypothetical melt is a slurry of liquid aluminum and aluminum alloy with solid particles of U_3O_8.

INTRODUCTION

These calculations were prepared in response to questions raised by the external reviewers of the HFIR Unirradiated Fuel Shipping Cask SARP concerning the assumptions that were made for the heat transfer calculations for the heating of the fuel elements in a fire. In order to resolve remaining questions, a bounding, worst-case scenario was devised in which the fuel element is completely ejected from the shipping cask and is exposed to the full intensity of the 1475° F (800°C) fire.

CALCULATIONS

The objective in this study was to do the most economical calculation that could be created while still being able to estimate the melting of the fuel. The model used is strictly one-dimensional with the effects of higher dimensions approximated conservatively. It assumed that, when any of the fuel or cladding metal melts, it will flow away from the melt surface. This is a conservative assumption because the molten material will require at least a small amount of time to flow away and will absorb heat during the time it remains on the surface. Another assumption is that the heat transfer occurs by thermal radiation from the fire to the molten surface, which is at the melt temperature of the material.

The configuration of the fuel elements is highly three-dimensional. However, the geometry of the fuel plates is such that most of the heat is applied to the edges of the plates. The plates are 50 mils wide with slots of the same width between them. Because the gap is so narrow, most of the radiant energy passing through the gap will fall on the surfaces near the edge of the plate. An approximation was made that all that radiant energy will fall on the edge, which means that the effective area of the edge of the plate will be doubled.

The melting temperature of 6061 aluminum (i.e., the outer cladding), as given in Table 3.2 of the SARP, is about 1080° F (582° C). The fuel is made from powders of U_3O_8 and 1100 aluminum. The U_3O_8 decomposes to UO_2 at 1300° C (2372° F), which in turn melts at 2500° C (4532° F). The 1100 aluminum, which is nearly pure Al should melt at near the melting temperature of pure Al, 1220° F (660° C)[1]. The worst case heat flux would be radiant transfer from the fire at 1475° F (~800° C) to the melting cladding at 1080° F (582° C). This flux would be

$$\frac{q}{A} = .1714 \times 10^{-8} \frac{Btu}{hr-f\ ^2-°R^4} / (1/.8+1/.9-1)\ [(1475+459.67)^4 - (1080+459.67)^4]$$

$$= 1.06 \times 10^4\ Btu/hr\text{-}ft^2$$

NOTE: The gray body emissivity for the fuel assembly as specified by the regulations is 0.8. However, data for even heavily oxidized aluminum surfaces give a value that is 0.3 or less[2], so that using a flux this high is quite conservative (if an emissivity of 0.3 is used in the above calculation, a heat flux which is about 40% of the above value results).

Using some data from Wendel and Morris,[1] it is estimated that the average specific heat for 6061 aluminum in the range from 169 to 1080°F is about 0.247 Btu/lb_m-°F. The heat rise from 169 to 1080 would be

$$(1080°F - 169°F)\ 0.247\ Btu/\ lb_m\text{-}°F = 225\ Btu/lb_m.$$

The heat of fusion of aluminum cladding is 171 Btu/ lb_m[1], so that the total heat needed to heat and melt the cladding is

$$\Delta h\ clad = 225 + 171 = 396\ Btu/lb_m$$

The above heat flux is applied to the surface with an aluminum density of 169 lb_m/ ft^3 [1], resulting in

$$\text{cladding erosion rate} = 1.06 \times 10^4\ Btu/hr\text{-}ft^2\ /\ 396\ Btu/\ lb_m\ /\ 169\ lb_m/ft^3$$

$$= 0.158\ ft/hr = 1.90\ in/hr$$

This is the nominal erosion rate into cladding material 6061 aluminum.

The plate is a sandwich of Al cladding surrounding the fuel. The thermal characteristics of the fuel are different from those in the cladding. The heat flux to the melting fuel would be that from 1475 to 1220°F, which is

$$\frac{q}{A} = .1714 \times 10^{-8} \frac{Btu}{hr-ft^2-°R^4} / (1/.8+1/.9-1)\ [(1475+459.67)^4 - (1220+459.67)^4]$$

$$= 0.76 \times 10^4\ Btu/hr\text{-}ft^2$$

According to Wendel and Morris,[1] the meat in the outer plates is about 71 wt% aluminum. The heat capacity from 169 to 1220°F (again, according to Wendel and Morris) is about 0.207 Btu/lb_m-°F. Therefore, the heat capacity for the heat rise from room temperature to melt plus the melt energy is

$$\Delta h\ outer\ meat = (1220°F - 169°F)\ .207\ Btu/lb_m\text{-}°F + .71\ wt\%\ Al \times 171\ Btu/\ lb_m$$

$$= 339\ Btu/lb_m$$

The average density of the meat in the outer plates is about 186 lb_m/ft^3.[3] The nominal erosion rate for the meat is outer meat erosion rate = .76x10^4 Btu/hr-ft^2 / 339 Btu/lbm / 186 lbm/ft^3

= 0.120 ft/hr = 1.44 in/hr

For the inner plates, the meat is about 77 wt% aluminum[1] with an average specific heat of .218 Btu/lb$_m$-°F [3] and an average density of 176 lbm/ft^3 [1]. Therefore, the heat capacity for the heat rise from room temperature to melt plus the melt energy is

Δh inner meat = (1220°F - 169°F) .218 Btu/lb$_m$-°F + .77 wt% Al x 171 Btu/ lb$_m$

= 361 Btu/lb$_m$

The nominal erosion rate for the inner plate meat is.

inner meat erosion rate = .76x10^4 Btu/hr-ft^2 / 361 Btu/lbm / 176 lbm/ft^3

= 0.119 ft/hr = 1.43 in/hr

As mentioned in a previous paragraph, the width of the area available for radiant transfer is twice that of the plate, so therefore the effective area is doubled. In the process of estimating an average erosion rate for the plates, it is important to look at the erosion of a sandwich whose component parts have different erosion rates and different melt temperatures. Erosion rates could conservatively be doubled. However, this would mean that the cladding erodes faster than the meat which is not consistent with the one dimensional, steady-state erosion model used for this problem. Therefore, it becomes important to visualize the configuration of the melting plate, and conclude that the cladding should melt a little further back than the meat. The result is that more of the surface area of the meat will be exposed to radiant heat transfer than the cladding. A stead state results when the two erosion rates are equal. Using the area multiplication factors an average erosion rate can be estimated. The heat flux for the cladding will be multiplied by one area weighting factor with that for the meat multiplied by a second factor. The sum of the factors will be 2.0. Requiring that the erosion rates be equal leads to the following equation for the cladding area weighting factor for the outer fuel, fout:

1.06x10^4 Btu/hr-ft^2 / 396 Btu/lbm / 169 lbm/ft^3 x fout =

.76x10^4 Btu/hr-ft^2 / 339 Btu/lbm / 186 lbm/ft^3 x (2 – fout)

which yields

fout = 0.864

and

outer plate erosion rate = 1.90 in/hr x 0.864 = 1.64 in/hr

The equation for the cladding area weighting factor for the inner fuel, finn, is

1.06x10^4 Btu/hr-ft^2 / 396 Btu/lbm / 169 lbm/ft^3 x finn =

$.76x10^4$ Btu/hr-ft^2 / 361 Btu/lbm / 176 lbm/ft^3 x (2 - finn)

which yields

finn = 0.860

and

inner plate erosion rate = 1.90 in/hr x 0.860 = 1.63 in/hr

The fuel plates are 24 in long. The fuel-bearing portion of the plates is 20 in. long. Figure 1 shows a schematic of the fuel plates as they are inserted into the side plates.

A schematic of an outer fuel plate drawn as if it were flat is shown in Fig. 2. The dimensions of the plate are given in the fabrication drawing, ref. 4, and those of the side plates in the outer fuel element drawing (Chapter 1, Fig. 1.8). The dimensions shown in fig. 2 are the averages of the maximum and minimum values.

A schematic of an inner fuel plate drawn as if it were flat is shown in Fig. 3. The dimensions of the plate are given in ref. 7 and those of the side plates in the inner fuel element drawing (Chapter 1, Fig. 1.7). The dimensions shown in fig. 3 are the averages of the maximum and minimum values. All dimensions are rounded to the even half mil (0.0005 in).

Before the heat can begin melting the fuel plates, it must melt through the outer side plate (cylinder). For this, the heat flux is not doubled because the plate is continuous. Thus, the appropriate erosion rate is the cladding value of 1.90 in/hr.

The side plate for the outer assembly is 0.128 in thick. The erosion of 0.128 in of aluminum side plate at 1.90 in/hr requires

0.128 in / 1.90 in/hr = 0.067 hr

In addition, the fire must erode a width of 0.0925 in of exposed plate which has no fuel before reaching the fuel-bearing portion. The appropriate erosion rate is the cladding value doubled, or 3.8 in/hr. This additional erosion requires

0.0925 in / 3.8 in/hr = 0.024 hr

For the outer plate, the time remaining for eroding the fuel-bearing portion until the fire is out is

0.5 - 0.067 - 0.024 = 0.41 hr

During this time interval, the erosion in the outer plates (using the 1.64 in/hr rate) will be

erosion outer element = 0.41 hr x 1.64 in/hr = 0.672 in

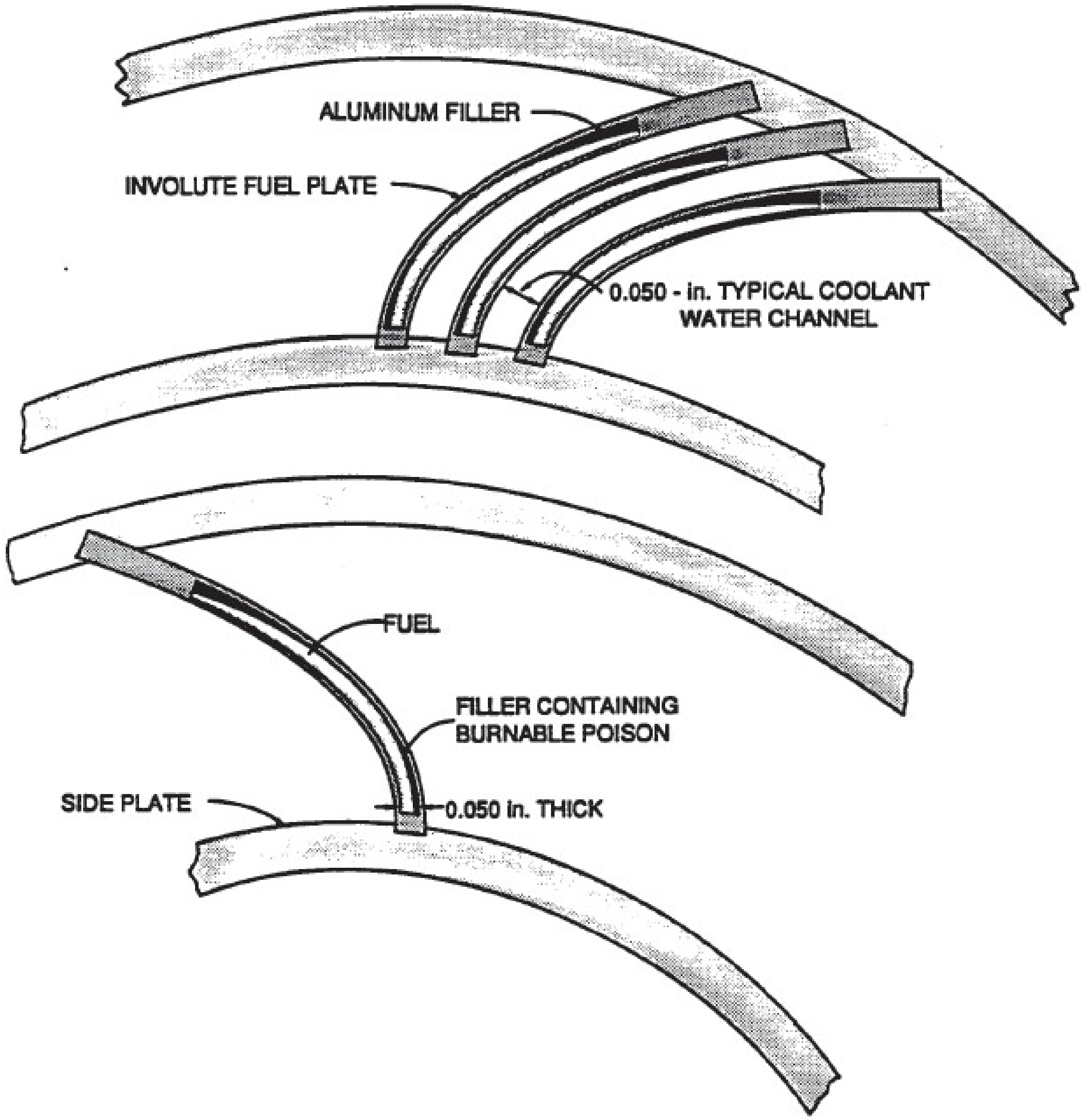

Fig. 1 Fuel Elements as Bent and Inserted Into Side Plates

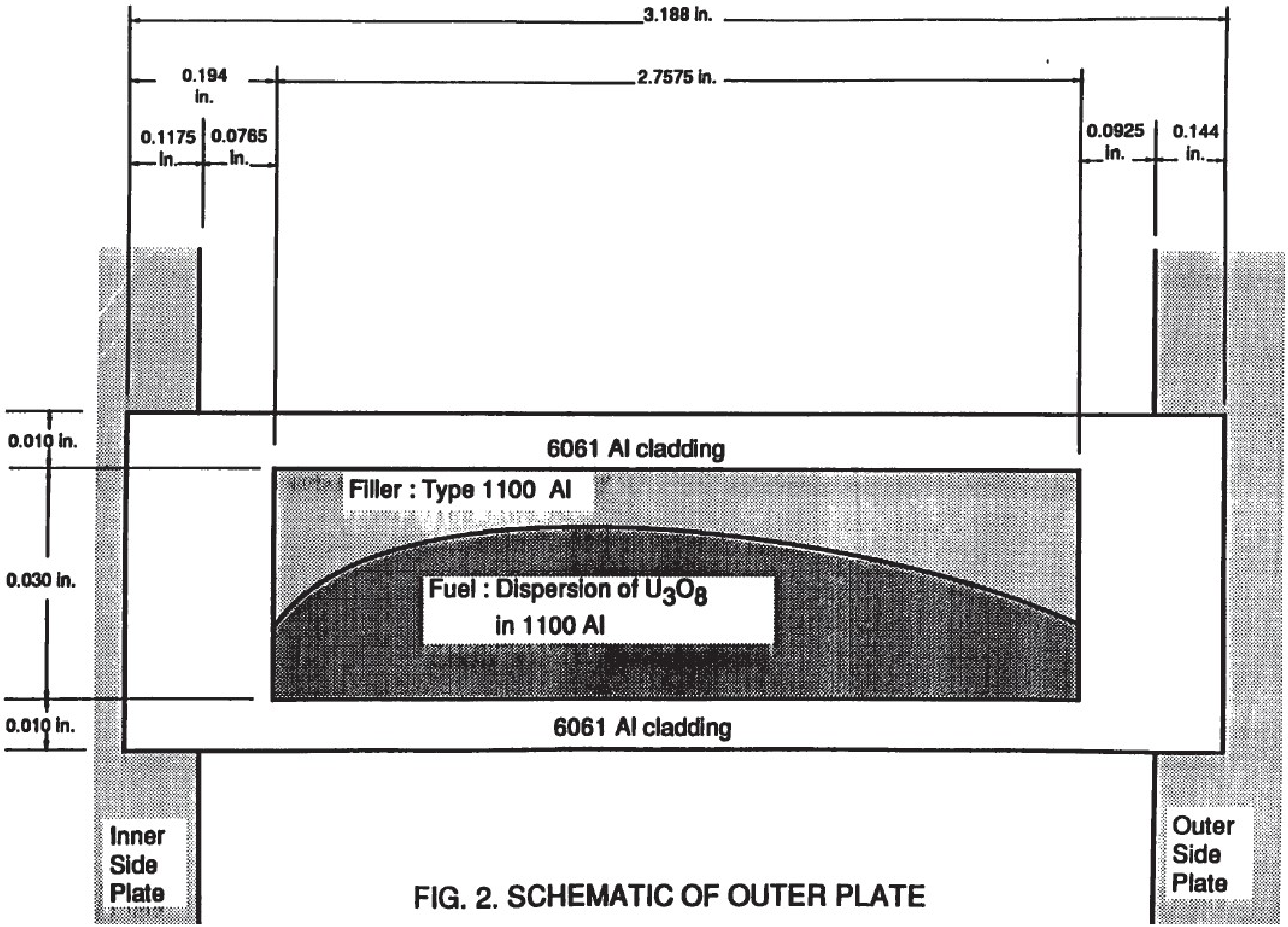

FIG. 2. SCHEMATIC OF OUTER PLATE

(Not to Scale)

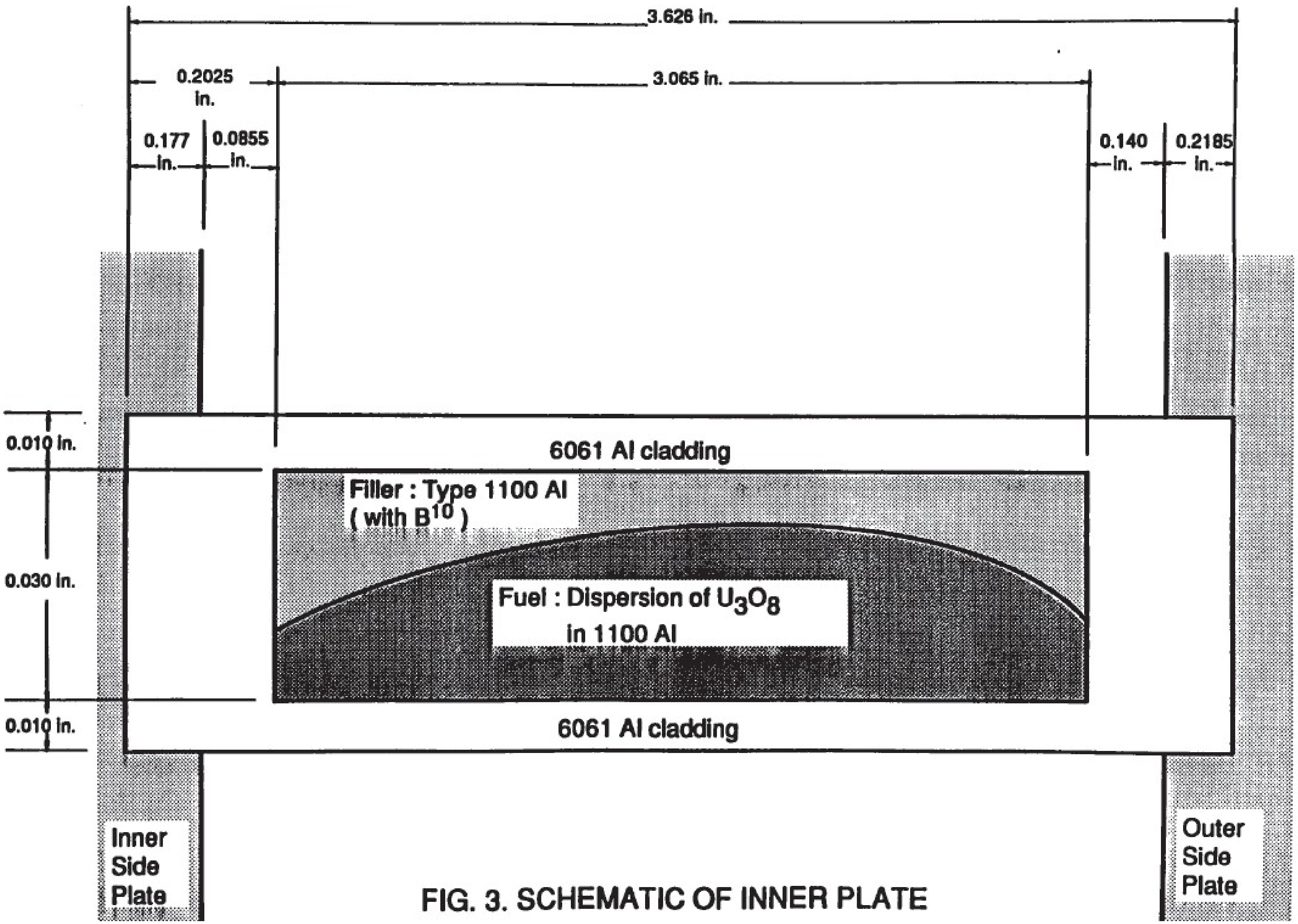

FIG. 3. SCHEMATIC OF INNER PLATE

(Not to Scale)

The side plate for the inner assembly is 0.105 in thick. The erosion of 0.105 in of aluminum side plate at 1.90 in/hr requires

0.105 in / 1.90 in/hr = 0.055 hr

In addition, the fire must erode a width of 0.140 in of exposed plate which has no fuel before reaching the fuel-bearing portion. The appropriate erosion rate is the cladding value doubled, or 3.8 in/hr. This additional erosion requires

0.140 in / 3.8 in/hr = 0.036 hr

For the inner plate, the time remaining for eroding the fuel -bearing portion until the fire is out is

0.5 - 0.055 - 0.036 = 0.41 hr

During this time interval, the erosion in the inner plates (using the 1.63 in/hr rate) will be

erosion inner element = 0.41 hr x 1.63 in/hr = 0.668 in

The erosion of the plates has been estimated. However, the ends of the plates are 2 in long, a section which has no fuel-bearing material. Thus, melting of the first 2 inches does not add to the hypothetical fuel melt.

The total length of fuel-bearing material for the outer assembly is

L total outer element = 2.7575 in

Thus, the fraction of fuel-bearing material melted from the outer assembly is

0.672 / 2.7575 = 24.4%

The total length of fuel-bearing material for the inner assembly is

L total inner element = 3.065 in

Thus, the fraction of the fuel-bearing material melted from the inner assembly is

0.668 / 3.065 = 21.8%

CONCLUSION

Based on the conservation evaluation above, less than 25% of the fuel bearing portion of a HFIR fuel assembly will melt if the assembly is exposed to a 1475°F (~800°C) fire for 30 minutes. The preceding analysis assumed an initial condition of 169ºF consistent with the maximum package NCT of 169ºF as identified in Chapter 3, Section 3.4.2.

REFERENCES

1. Wendel, M. W., and D. G. Morris, *High Flux Isotope Reactor System RELAP5 Input Model*, Report ORNL/TM-11647, Draft 12-20-90.

2. *Standard Handbook for Mechanical Engineers*, T. Baumeister and L.S. Marks, eds., McGraw-Hill, New York, 1967.

3. W.H. McAdams, *Heat Transmission*, McGraw Hill, New York, 1954.

4. "Fuel Plate Outer Flat (Enriched)", Drawing 8-7148, M&C Nuclear, Inc., Attleboro, MA.

5. Deleted.

6. Deleted.

7. "Fuel Plate Inner Flat (Enriched)", Drawing 8-7146, M&C Nuclear, Inc., Attleboro, MA.

8. Deleted.

9. Deleted.

been demonstrated to be essentially impermeable to fission product migration or diffusion below 500° C (932° F).

The HFIR fuel is highly enriched Uranium (U-235 specifically addressed as 93.164%) and therefore is characterized as a mixture of isotopes to determine the A2 value for the mixture. The following data provides the characterization of the HFIR fuel material in determining the A2 value per unit mass.

Isotope	Specific Activity (Ci/g) (1)	Weight %(2)	Ci/g-U	Fraction of activity
U-234	6.2E-03	1.010	6.26E-05	0.9643
U-235	2.2E-06	93.164	2.05E-06	0.0316
U-236	6.5E-05	0.389	2.53E-07	0.0039
U-238	3.4E-07	5.437	1.85E-08	0.0003
U Fuel		100	6.49E-05	1.0000

(1) 10CFR71 Appendix A, Table A-1
(2) Contents as listed in Chapter 1, page 1-4

Appendix A is used to determine the A_2 value for the fuel. Note that the 10 CFR 71, Appendix A, indicates the A_2 values for U-235 and U-238 are unlimited. The A2 for the uranium fuel is determined by

$$A_2 \text{ for mixture} = \frac{1}{\sum_i \frac{f(i)}{A_2(i)}}$$

where f(i) is the fraction of activity for radionuclide i in the mixture and $A_2(i)$ is the appropriate A_2 value for radionuclide i. The A2 values for U-234 and U-236 are both 1.6E-01 Ci based on the slow lung absorption values provided in 10 CFR 71, Appendix A . The determined A2 for the uranium fuel mixture and uranium fuel activity for inner and outer fuel elements follows

$$\text{A2 for mixture} = 1 \div ((.9643 + .0039) \div 1.6E-01) = \mathbf{1.65E-01\ Ci}$$

$$Activity\ of\ Uranium\ for\ inner\ element = 2630\ g\ (U235) * 6.49E-5\ Ci/g \div 93.164\% = 1.83E-1\ Ci$$

$$Activity\ of\ Uranium\ for\ outer\ element = 6880\ g\ (U235) * 6.49E-5\ Ci/g \div 93.164\% = 4.79E-1\ Ci$$

Therefore, an amount of fuel-bearing material exceeding 90 percent of an inner element or exceeding 34 percent of the uranium in an outer element constitutes a quantity exceeding the A2 value.

A secondary boundary is formed by the container cavity, lid, and seal as indicated in Fig. 1.1 (outer HFIR element shipping container) and Fig. 1.2 (inner HFIR element shipping container). This boundary, which surrounds the containment or fuel cladding, is necessary only for ensuring compliance with the structural analysis. The integrity of this secondary boundary establishes the cavity conditions for the fuel cladding (containment).

week is allowed. To release an A_2 amount of material the following conditions would have to occur.

Inner container – Release of greater than 90% of fuel from the cladding/containment

Outer container – Release of greater than 34% of fuel from the cladding/containment

A bounding worst-case thermal analysis for the hypothetical thermal event was also performed that assumed no barrier protection was offered by the shipping container. This conservative analysis (Appendix B of Chapter 3) concluded that less than 25% of the fuel-bearing material (U_3O_8) of the outer or inner fuel element will melt when only the fuel element is subjected to the hypothetical thermal event.

As shown in Chapter 2, no significant damage is experienced by the fuel element from the 30-ft drop test analysis. Therefore, the containment feature of the fuel as outlined in Sects. 4.1 and 4.2 in association with the shipping container demonstrates that less than an A_2 value of radioactive material is released in the hypothetical accident conditions.

The test on irradiated fuel plates documented in Appendixes A and B are based on miniature fuel plates fabricated according to the requirements outlined in Appendix D. These fuel plates were fabricated and accepted according to the following minimum criteria:

Dimensional inspection,

Blister inspection, and

Radiographs of each plate

Because these tests are encompassed in the verification/acceptance tests used in the standard fabrication of fuel plates contained in the fuel element shipped in this container as discussed in Sect. 4.1.4, more than adequate confidence exists that containment is established for these fuel plates when they are subjected to a hypothetical thermal event. A bounding thermal analysis performed in Chapter 3 showed that when no container is present, less than an A_2 quantity of material will be released. Therefore, when the hypothetical thermal event is imposed after the drop test, no release of material greater than an A_2 value will occur.

Several irradiated fuel plates were bent to determine their ductility and toughness as described in Reference 5. These tests were performed in 1986 as part of the RERTR Program. These tests are conservative because unirradiated aluminum is more ductile than irradiated material. Figure 4.2 is a photograph of three plates after bending to evaluate toughness. These tests documented that when plates are subjected to greater than a 90° bend the bonds between the cladding and the fuel will not break. Therefore, containment is conservatively maintained when the plates are bent to angles less than 90°. These test plates were fabricated according to a process similar to that used for the fuel plates used in the HFIR fuel elements. Because the dimensions of these irradiated fuel plates (worst-case condition) are similar to the HFIR fuel plates, no release of radioactive material will occur when unirradiated fuel plates are bent to angles less than 90°. The structural calculations in Chapter 2 show that plate bending is limited to less than 10°.

In summary, because of the extent of fuel melt required (>34% for outer fuel element and >90% for inner fuel element), there is no likelihood that radioactive material in excess of A_2 quantities will be released from this package as required per the applicable regulations.

Therefore, fuel plates used in the HFIR fuel elements provide the necessary containment to meet the regulatory requirements for both normal transport and the required hypothetical accident conditions.

4.4 Special Requirements

No plutonium shipments are made in this shipping container. Therefore, this section does not apply.

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SAFETY ANALYSIS REPORT FOR PACKAGING THE ORNL HFIR UNIRRADIATED FUEL ELEMENT SHIPPING CONTAINER VOLUME 2

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